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They're Off . . .

. . . at least not quite. We have not yet got to the stage of depressing the clutch, engaging first gear, lightly pressing the accelerator whilst at the same time slowly releasing the clutch pedal. Nor for that matter has the motor actually been revved very hard, but at both Berkeley and Bradwell it has been started and furthermore it seems to be operating in much the same manner that the designers intended it to. It would appear also that the test teams have avoided some of the more obvious traps for young engineers such as, confusing the radiator system with the inlet manifold and getting water into the cylinders (Windscale) or for that matter filling the wrong sump with oil (Calder). Indeed the heavy air of quiet confidence is something of an anticlimax to the trials and tribulations of the past five years.

But after four reactors at Calder and another four at Chapelcross have gone smoothly into operation it would have been more surprising if the sums done on Berkeley and Bradwell had not been proved reasonably accurate. Seven of the Calder/Chapelcross series were, of course, repeat performances and could be expected therefore to follow a predetermined pattern while both Berkeley and Bradwell are quite different and quite a lot bigger than their Calder counterparts. Also so many modifications have been made to the designs of the cores and they are now so much more complex than the simple graphite stacks of yesteryear, what with zirconium pinning and the like, that some introduction of errors was forgivable. However, from a reactivity point of view all seems to be well and we must now wait and see how the reactors and their auxiliary circuits behave as power is increased. This will not be an immediate event because an extensive programme of zero energy testing has been drawn up, flux plotting, fine structure measurements and so on, so that full advantage can be taken of the relative inactivity of the system, a condition that will never again obtain.

The start-up of Berkeley and Bradwell has seen the Minister of Power's inspectorate for licensing and insurance assume active responsibility for the first time. This is the organization that has the power to say yea or nay to any operational procedure and has the last say as to when an installation is ready to go. The importance of this body's activities to the general public is, of course, very great; it is the men from the Ministry of Power who are entrusted by the public with the task of ensuring their safety and to seeing their lives are not menaced by this new industrial hazard. There is, of course, no reason why the operation of nuclear power stations should not always be as safe and as harmless to the general public as the operation of conventional stations. Indeed one could go much further than this and say that there is every reason why a clean, compact, nuclear generating station should not be much less of a menace to the public at large than its equivalent coal or oil fired station liberating tons of harmful chemicals into the atmosphere, continuously, and at all stages of its life. But this will only be so if the appropriate precautions are taken and if the procedures adopted are soundly based. The Minister's men therefore are the public's key representatives in this field, with a duty to ensure a clean record for all time.

Even remembering Windscale the record in this country of reactor operation is excellent (and due credit must be given to the AEA who have had the difficult job of designing, judging and operating, always without the benefit of outside criticism). It is strange, therefore, that this new body should have begun its operations in a manner which suggests that it has inherited a house full of cupboards each one of which has at least one skeleton in it. Far from behaving as if they were representing the public at large the Ministry's attitude towards questions regarding Berkeley and Bradwell was initially suggestive of a secret society operating in some life or death military exercise. In contrast the CEGB, whose practice in regard to conventional stations is to say as little as possible until everything is fully operational, have been most co-operative and understanding of the keen interest surrounding this first commercial exercise into criticality.

The Minister would do well to study the record of the AEA's relations with outsiders. The Authority was initially created for the prime purpose of establishing a war-potential and ever since has been shackled with the traditions of the (at first necessary) secrecy that had to be imposed. We have pointed out on many occasions that the decreasing emphasis on military work has not led to open house but that lack of candour has persisted in the guise of commercial security, confidential work, consultancy and so on. We believe that this is now in most instances contrary to the policy of the members of the Authority; and they are sincere in their desire to classify secret only those things that are definitely of military significance or indisputably of material commercial value. The snag is that so deep seated are the old traditions, the change in attitude is not easy to organize down through the ranks and on the other side, so strained can relations with the outside world become that even when complete frankness is the order of the day the "official spokesman" becomes the last to be believed. The position is steadily improving with the Authority; it is far from ideal but at least the Authority is less and less inclined to believe the worst of any enquirer's intentions and news media are less inclined to convert any minor item of passing interest into a major scandal.

The moral is clear; if the Ministry of Power inspectorate wishes to avoid acquiring a reputation for dissembling then it must not treat enquiries regarding its *modus operandi* and technical questions relating to the safety of

the plants that it supervises as fifth column activities aimed at sabotaging the nuclear programme. It must be prepared to recognize the genuine interest (and the anxiety) that surrounds its activities. Probably at the moment it is simply a matter of organization and the communication machinery is not yet properly commissioned. The danger is that unless proper importance is attached to the inspectorate's public relations from the outset, a wall of distrust will be built up leading to a barrage of questions (both inside and outside the House of Commons) which become more and more "loaded" as time goes on until the statement of the "official spokesman" is always disbelieved. We believe that should this happen, a great disservice would be done to the public and the nuclear engineering industry and the work of the Ministry of Power would be made much more difficult without any compensating advantage. Like justice, it is not sufficient for safety to be achieved, it must also be *seen* to be achieved.

The Future of the IAEA

IN a few days' time the annual general conference of the IAEA will be convened in Vienna. No doubt the opening sessions will be devoted to discussions as to whether Red China should be admitted, whether there should be a triumvirate to supplant the director-general and similar political wrangles which have been aired elsewhere and have produced exactly the same amount of ill-feeling without any concrete results. The Annual Report for 1960-61 which the IAEA has published of its activities shows something of the work the organization can do in terms of assistance to underdeveloped countries in the way of advice, training of staff, awarding of fellowships, etc. But it is made plain that the work of the Agency at the moment is seriously handicapped by a shortage of funds and this in spite of the fact that the General Assembly of the United Nations has urged that greater funds should be made available.

It is obvious that the right and left hands of the member nations are not *en rapport* and the IAEA would do well at this September meeting, instead of arguing about points of procedure, to consider seriously whether the IAEA is to be made a practical useful force, or simply another platform for airing political views and wasting VIPs' time, or whether it were not better to call it all off here and now. It is ridiculous to form an International Agency and then not provide it with the funds that it requires for its operation. Its collaborative work on safety standards and the like is undeniably of great importance but do the nations now feel that much of its work is unnecessarily high-powered? Study teams of international standing are not set up to recommend what type of electric locomotives should be bought by a new community, whether it should go in for front wheel or rear wheel drive motorcars, what form and what type of aeroplane should be used for its airlines, so why is it necessary for nuclear power? If this is the general feeling then let it be said honestly and not simply a financial squeeze applied.

But for the generous contributions of the U.S. the IAEA would almost cease to exist overnight, so this year let us get down to fundamentals and determine, if the IAEA is to continue, exactly what its functions are to be and then abolish the voluntary fund so that the permanent secretariat can get down to doing its allocated job knowing from the outset what funds it has available.

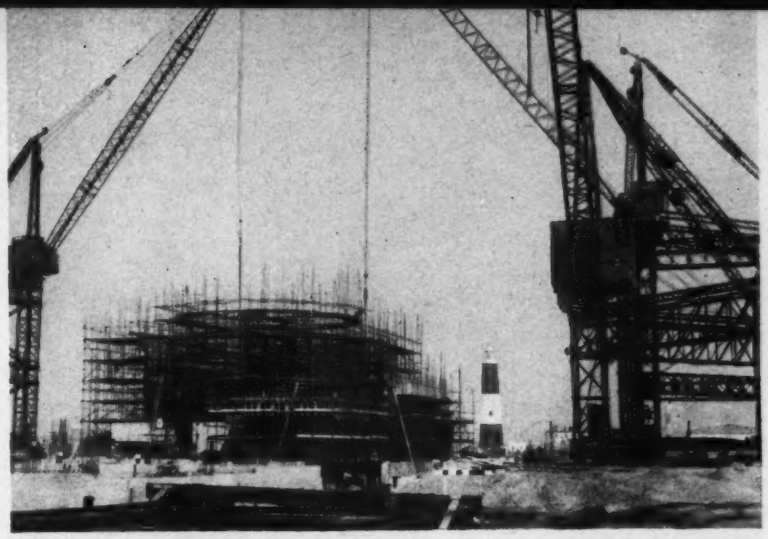
The PWR Family

IN this edition of *Nuclear Engineering* we feature *in extenso* another in the line of U.S. pressurized water reactors, Indian Point, which Consolidated Edison are completing north of New York and for which the nuclear design has been prepared and executed by Babcock and Wilcox. Indian Point will be the third large land-based PWR to come on stream, its predecessors being Shippingport and Yankee. In smaller sizes there are already many examples of PWR power reactors in operation, for example, the U.S. submarine fleet, BR-3 in Belgium, the Greenland and Alaskan remotely sited plants, and in the medium-power range, the three Russian PWRs driving the *Lenin* and the PWR in the N.S. *Savannah*. The PWR is the oldest power reactor concept (or perhaps more accurately it should be said, has been under development for a longer period of time specifically for the generation of power than any other system) and its obsolescence has been forecast at regular intervals. It seems, however, likely to remain in the forefront of interest for many years to come. The largest single reactor unit at present proposed is in fact a PWR which Westinghouse would like to build for Southern California if siting problems can be resolved. This would be a 360 MW plant, a single reactor unit apparently without the additional benefits of any fossil fired superheating. A Babcock variant of the PWR that is exciting interest is the spectral shift reactor which may again put off the day of obsolescence for some considerable time.

Two aspects of the Indian Point design distinguish it from its predecessors; one is the use of thorium in the core (in sufficient concentration for conversion to U^{233} and its subsequent burning to be significant) and the second is the provision of an oil fired superheating plant to improve steam conditions. Measurements made over the past two years have considerably narrowed the margin of uncertainty in the physics properties of U^{233} but Indian Point will still contain the first core of any size operating at power in which U^{233} will make an important contribution to the fissioning process.

On the superheating question there is still a great deal of controversy and passions are likely to be raised and arguments used which are not particularly soundly based. So far as Consolidated Edison is concerned (and the same will be true for the purchase of steam from Elk River) the investment in this additional plant will pay for itself hands down. The purists protest that if oil fired superheating brings economies then a completely oil fired plant would have been more economic. At present this is of course true, but it would be a mistake to assume that it is a truism any more than, say, mixed forms of transport are automatically more economic than a single system. Far then from denigrating this particular exercise its results should be examined with care for, even in the long term there may be a future for fossil fired superheating in certain special cases.

World Digest



No. 1 reactor for Dungeness nuclear power station under construction by The Nuclear Power Group for the CEGB.

International

A two-year extension to the U.S.-Soviet nuclear energy exchange agreement is being sought. Although implementation of this agreement, signed in 1959, has to date been "only moderately successful," it is felt that it is nevertheless worth continuing. Greatest success has been in the realm of personnel exchange, although even here everything has not gone as smoothly as hoped. Agreement has not been possible, for instance, on what power plants should be visited. Exchange of information has been beneficial and some sort of progress has been made on the proposals for constructing a joint accelerator project.

The IAEA's fifth annual general conference opens on September 26 in the Hofburg, Vienna. It is expected to last for two weeks, the first of which will be devoted to a general debate. It will not be unexpected if the various representatives will again use the conference as a political platform, and urge a reform in the overall organization of the Agency.

The ENEA Ship Propulsion Study Group meeting scheduled for September 15 has been postponed until October 5. The group is to receive reports from its working parties (*Nuclear Engineering*, September, 1961).

Samples of Am^{243}, Cm^{244}, Bk^{249} and Cf^{252} have been supplied by the AEC to five European nuclear centres for research purposes. A charge of $100 has been made for each sample and the recipients guarantee to make available to the U.S. the results of work carried out using the samples. The organizations supplied are CEN, at Mol, Belgium; the CEA, France; Institute for Physical Chemistry, Bonn University, on behalf of Kernreaktor, Karlsruhe; the Institutt for Atomenergi, Kjeller, Norway; and the Research Institute for National Defence, Stockholm.

Fifty-six proposals for joint research and development work between the United States and Euratom have been authorized since the start of the collaborative programme in December, 1958, according to a recent report. Most of the proposals are now under contract, a few have in fact been completed and the remainder will be taken up in the near future. Twenty-eight Euratom and 11 U.S. organizations are involved in the work. The U.S.-Euratom Joint Research and Development Board also report that the first of a series of quarterly summary digests of the work in progress under the joint programme is now available.

Fast reactors as practical, economic power-producing machines of the future were coming into prominence, said Dr. D. B. Hall, of Los Alamos Laboratory, summing up the recent IAEA seminar on the physics of fast and intermediate reactors. There was, however, he added, a long way to go to attain the same degree of facility in the calculation and design that exist with the thermal reactors. Altogether 74 papers were presented at the seminar which was attended by 133 scientists from 22 countries and two international organizations.

A provisional budget of $500 million to cover the Euratom research and development programme for the next five years was approved by the French and opposed by the German and Belgian delegations at the meeting of the research advisory committee on September 13-14. Opposition was made solely for reasons of reducing national expenditures. It is believed the French have particular interest in maintaining the original proposed level as they are hoping for Euratom support in their fast reactor programme. The budget would, of course, be increased if Britain were to join Euratom and it might well be with this in mind that Germany and Belgium are not over reluctant in requesting at this stage a budget reduction. At the same meeting the committee set up three sub-committees for dealing with scientific problems.

What was called an almost complete up-to-date picture of work throughout the world on controlled nuclear fusion was presented during the week-long IAEA sponsored conference on Plasma Physics and Controlled Nuclear Fusion Research at Salzburg during early September. Over 500 delegates from 29 countries and six international organizations attended, and 109 papers were presented. In addition to the 10 scheduled sessions there were also 14 informal sessions attended on an average by 70 participants.

United Kingdom

Permission is sought by the CEGB for construction of an oil-fired power station at Fawley, on Southampton Water. If obtained the plans for a nuclear power station at Hamstead, on the Isle of Wight, will be shelved. Fawley, suitable for a 200 MW plant, is close to the Esso oil refinery and there would be few problems, if any, about fuel supplies. First generating unit should be ready by 1967. The idea of a nuclear power station at Hamstead was mooted in 1959 after the Board had surveyed the coast between Dungeness and Weymouth for suitable nuclear station sites. Only two were, in fact, found, the Hamstead one and another at Earnley, Sussex. Detailed investigation followed and at the end of last year the Board declared Earnley unsuitable but Hamstead excellent. However, because of the amenity value of the area the Board decided to defer any decision about using the site until alternative proposals for building conventional plant had been investigated.

Rutherford College of Technology, Newcastle-upon-Tyne, is to install a sub critical assembly for use by its physics department. The assembly, to use natural uranium and light water, is being designed by college staff and components for it will be built either in the college workshops or by local industry. It will be installed in a part of the new Rutherford building (named incidentally, not after the physicist but a local benefactor, who also came from the Antipodes).

IAEA's Fourth Annual Report

Overall the IAEA's fourth annual report reads as a record of consolidation and some sort of achievement. There is no escape, however, from the sense of all not being well. Some member states are not contributing enough in the way of finance and as a result the Agency is obliged to make reductions in the programmes concerning fellowships, the provision of experts and equipment, research contracts, the use of the mobile isotope laboratories and contributions to the Agency's laboratory at Seibersdorf. As an example of the effect of this, the programme for technical assistance for 1961-62 has had to be curtailed by some 60%, while under the fellowship programme only 300 fellowships will be awarded, compared with 429 in 1960-61. Assessed contributions for 1960-61 from member states amounted to $5 880 980: by December 31, $5 312 034 had been received; by June 30, 1961, $5 441 500. For 1961-62 the total contributions due were assessed at $6 195 756, of which 41·52% had been paid by June 30, 1961. During the year ended June 30, 1961, which is covered by the report, the Agency awarded 48 new research contracts, renewed 35 others and spent $652 189 on them; organized 13 meetings on different subjects which were attended by 2 327 participants, and at which 762 papers were presented; and issued 50 scientific and technical publications. The report will be considered at the Agency's annual general assembly in Vienna opening on September 26.

The Royal Naval College hopes to have the Jason from Langley installed and in operation at Greenwich by June of next year. A contract covering the transfer has yet to be signed but negotiations with AEI, who are acting as contractors in the project, are progressing smoothly. As an Admiralty project the installation and operation of the reactor would not come under the jurisdiction of the Ministry of Power. Nevertheless, in order to allay any doubts that might arise in the public mind, the college's proposals have been submitted to the Ministry's Inspectorate of Nuclear Installations for scrutiny and, it is reported, approved.

During a thunderstorm on September 2 lightning struck a grid line serving the AEA at Calder and Windscale, resulting in a voltage drop on the blowers to the four Calder reactors. Three of the reactors were automatically tripped out of operation; the fourth, operating at a lower rating than the others, continued in operation, providing power for the works generally and to the Cumberland supply grid. Operation of the other three reactors was resumed the following morning. Although not a specifically nuclear incident, it did serve to demonstrate the efficacy of Calder Hall's automatic tripping system.

Permission to proceed with the loading of Bradwell No. 1 reactor following the successful completion of criticality tests was given by the Ministry of Power on September 8. Two thousand channels out of the possible 2 564 were to be loaded, following which neutron flux measurements and various tests would be made. Control rod calibrations would then be made, another permit probably being required for this, after which the BSD equipment would be tested. Further Ministry of Power permission will be required before operation of the reactor at any steady power level can begin. A report on the loading to criticality stages and tests together with a description of the part played by the Ministry of Power in these operations appears on page 405 of this issue.

About a dozen science masters and teachers from public and grammar schools throughout the U.K. attended a five-day summer course arranged for them by the AEA at Windscale and Calder, during August. They were instructed in the fundamentals of reactor design and operation, given the opportunity of using the Calder Operation School simulator and shown over the plant.

The linear accelerator injector which forms the first stage of Nimrod, the 7 GeV proton synchrotron under construction at Harwell went into operation at full power for the first time on August 1, producing a 15 MeV proton beam. Commissioning of the injector started a month earlier. The whole of the synchrotron is expected to be completed during 1962.

A 4GeV Van de Graaff machine has been ordered by Smith and Nephew, the surgical equipment and materials manufacturers, for carrying out research on radiation sterilization. It is the first step in a programme which it is expected also to cover the use of cobalt-60. The accelerator is scheduled to be in operation by the autumn, 1962.

Four hundred physicists from 30 countries attended the Rutherford Jubilee international conference held at Manchester University, September 4-8. Among them were (*left to right*) Sir James Chadwick, Sir Charles Darwin, Sir Ernest Marsden, Professor E.N.da C. Andrade and Professor Nils Bohr.

For about £3½ million the National Institute for Research in Nuclear Science is to install an "Atlas" electronic digital computer at Harwell for use not only for itself but by universities, government departments and the AEA. It should be ready for operation by 1964. The "Atlas," made by Ferranti, has been developed in co-operation with Manchester University, where the prototype is now being assembled. It is reputed to be the world's most powerful machine of its kind.

From Hunterston it is reported that pressure vessel for reactor A has been successfully stress relieved; the eight boilers have been set up and tubing of four of them is progressing well; installation of the first and second turbo-generators is on schedule and the switchgear in the turbine hall is getting on well. Bedevilled in the past by strikes construction of the station is now progressing steadily, although GEC, main contractors for SSEB, are reluctant to name any projected completion date for the whole station. Apparently negotiations between them and various organizations concerning the programme have not yet been concluded.

Australia

Lucas Heights has made its first delivery of cobalt 60 to an Australian hospital for use in treating cancer. It went to St. Vincent's Hospital, Sydney, which has previously had to rely on supplies from Canada.

France

Euratom co-operation in the fast reactor programme has been forecast by Dr. G. E. Vendryes, of the CEA, speaking at the recent IAEA seminar on fast reactor physics. This could well mean that France will for the first time seek financial aid for a specific large-scale reactor project. Apart from the Rapsodie, now under construction at Cadarache, the fast reactor programme allows for a 250 MW (th) reactor to be built before 1970. To date France has opposed the apportionment of Euratom money to specific large reactor projects on the grounds that they have not been "research" projects. The proposed successor to Rapsodie could certainly not be barred on these grounds.

West Germany

Latest company to announce financial support from the Bundes Ministry of Atomic Energy is Interatom, of Cologne, who have been guaranteed DM18 million for a sodium cooled reactor project using zirconium hydride as moderator and uranium carbide as fuel. Total cost of the design development and experimental stages for this project is put at DM25 million. It is understood the Jülich nuclear centre will co-operate with the company in some of the early experimental work. Government support for other major reactor projects announced recently have been DM18 million for the BBC-Krupp HTGR project, DM16 million for the AEG superheat BWR, and DM18 million for the B and W GCR.

A 0·1 W(th) teaching reactor is to be installed alongside the Bavarian College Argonaut at Garching, near Munich. It will be supplied by Siemens.

A second core is planned for the GKSS pool reactor at Hamburg to be used in conjunction with the first and so reduce the need for changing its position in the pool for different experiments. There are four core positions but at the moment only two of them are used with any frequency.

An active effluent storage tank is being built at Geesthact/Hamburg by GKSS in collaboration with the four northern German state authorities. Measuring 10 m × 30 m, it is intended to be completed before the end of this year, when the first fuel discharge from the GKSS reactor is expected.

India

Seven tenders have been received by the Department of Atomic Energy for the proposed 300 MW Tarapur nuclear power station. They come from the U.K., U.S., France and Canada. Evaluation of the tenders is expected to take five months and construction work started on site in March, 1962. United Kingdom companies known to have submitted tenders are GEC and the English Electric, Babcock and Wilcox, Taylor Woodrow Atomic Energy Construction Co.

Construction of the heavy water plant at Nangal, India's first, is scheduled to be complete by November. An adjunct to the existing fertilizer plant, it has a design output of 14 tonnes of heavy water a year. Half of this will be bought by the Atomic Energy Department and the remainder may initially be exported. Cost of production at the Nangal plant is expected to be about £10/lb.

The state government of Madras is considering the construction of a nuclear power station and may provide Rs1 million (about £75 000) towards a preliminary investigation.

Italy

A preliminary agreement has been signed by Simea with the U.K. AEA for the supply of 400 tonnes of uranium fuel for Latina at an estimated cost of £8 million. Before negotiating with the AEA Simea is understood to have sought other sources but nobody could offer buy back guarantees similar to those made by the Authority. As the Latina core loading amounts to about 262 tonnes it would appear from the preliminary agreement that Simea hold little hope for even a second core to come from elsewhere.

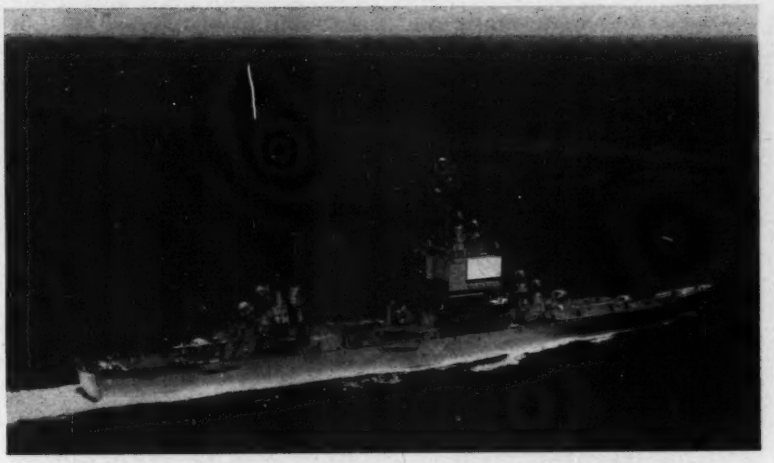
Netherlands

Work is in hand on the design of a 1 kW critical assembly for studying reactor core physics at Ispra, Italy, for the Euratom Orgel project. Known as Eco, the assembly is being designed by Neratoom in conjunction with the engineers, Comprimo, engineering consultants, Tebodin, and Reactor Centrum Nederland. Construction of Eco, which will use heavy water as moderator, is expected to start later this year. Criticality should be reached some time in 1962.

Norway

Conversion of the BHWR, Halden, into a spectral shift controlled reactor following the completion of the OECD programme either in 1962 or 1963, is proposed by the Institutt for Atomenergi. They are interested in this concept for marine applications and have in fact carried out both economic and design studies for a nuclear-powered oceanographic research vessel based on this concept. If modified the BHWR would require a new sub core and, to obtain a better stability margin for the system, a new subcooler loop. The primary pressure would be kept at 40 kg/cm² and the total reactor output at 20 MW. The subcooler would be designed for removing up to 50% of the reactor power, as against 15% that can be removed by the present loop.

An oceanographic research vessel has been proposed as a suitable subject for Norway's excursion into nuclear marine propulsion. In an Institutt for Atomenergi report it is said that from an economic point of view the construction and operation of such a vessel would bridge the gap between conventional and nuclear shipping and at the same time it would not be impaired in its operation by restrictions on entering foreign ports for refuelling. The reactor used would be based on the spectral shift control concept. Studies for the vessel, which are being carried out in conjunction with the Oceanographic Research Institute, Bergen, and the Ship Technical Research Institute, Trondheim, are still in a very early stage.



Put into commission on September 9 was the U.S. Navy's first nuclear propelled cruiser, *Long Beach*, seen here during her sea trials. Powered by two PWR units designed and built by Westinghouse, she is capable of exceeding her design speed of 30-31 knots. Equipped with the latest warfare instrumentation, she has been described as the Navy's most complicated ship. She must also be the most expensive of her type, estimated cost being $320 547 000, of which $41 million was spent on the reactor plant.

Portugal

Portugal is the twelfth country to receive a reactor grant from the United States. A cheque for $350 000 in respect of the AMF pool type reactor installed at the Junta de Energia Nuclear laboratory, Scaven, near Lisbon, was formally presented to the Junta president Dr. J. F. Ulrich, on August 28.

Sweden

The future of the proposed 100 MW(e) Marviken nuclear power plant, to be built for the State Power Board by Nohab and Asea from reactor designs by AB Atomenergi seems at the moment to be uncertain. The Power Board, who have been involved in the preliminary design stages since 1959, have announced that they will only continue in the project if the Government will free them from payment of interest on the principal investment. They feel, economically, there is no justification for them to build the station simply to gain experience. AB Atomenergi, however, believe such a station to be a necessary step in Sweden's nuclear power programme and will probably press the Government to provide appropriate aid. Completion of the station has been provisionally scheduled for 1967.

U.S.S.R.

Cancellation of a 50 MW(e) prototype fast reactor project was revealed at the recent IAEA conference on fast reactors by Dr. Ivan Bondarenko. It is believed to be the fast sodium cooled breeder based on a Pu^{239}-U^{238} cycle that was planned to be built at Ulyansk.

IAEA Report on Philippines

NUCLEAR power plant installed during the late 1960s in Luzon, the largest island of the Philippines, might be economically competitive over its lifetime with conventional power plant of the same size, according to a recent IAEA report. Prepared by a reactor engineer and two economists, the report analyses the power situation on the island and estimates growth in its demand. It is concluded that the growth is such to justify the installation of a large plant consisting of either one 150 MW or two 100 MW units connected to the grid towards the end of the decade. The three-man team further found, basing their calculations on the present price for oil at about 38 c/10^6Btu, that the use of nuclear power in either 2x100 MW or 1x150 MW units deserved serious consideration. Installed capacity in Luzon at the end of 1960 amounted to 489 MW, produced about half and half by the National Power and Manila Electric corporations. Of this total 287 MW is supplied to the grid, the rest going to industrial and other large scale consumers. Total energy demand in 1960 is given as 1 854 kWh, by 1970 5 350 kWh, the peak demand in these years in MW being 380 and 1 193, respectively. About half of the present installed capacity is from hydro plant, the rest from oil fired stations. Although there is an estimated potential of 2 000 MW to be obtained by hydro means the lack of information on rivers, etc., makes any definite forecast impossible. Other possible power sources seem limited and the choice for future requirements has to be made between oil and nuclear power. The IAEA has indicated its availability in assisting in any nuclear power projects the Government might decide to embark upon. It is believed in fact that the Government will be approaching the Agency for aid for such a project either later this year or in 1962.

U.S.A.

The Advisory Committee on Reactor Safeguards (ACRS) has reviewed information on the crazed cladding of the 22 MW(e) Elk River BWR pressure vessel and found it meets with the minimum safety requirements. It recommends that further reviewing is unnecessary before permission to operate the reactor is given. Consideration of issuing an operating permit was deferred earlier this year when the AEC regulatory staff requested further information on cracks that had been detected in the stainless steel cladding, a request later repeated by the ACRS.

Before granting Consolidated Edison permission to operate their 225 MW(e) combined nuclear reactor/conventional superheater plant at Indian Point, the AEC require additional information on the metallurgical characteristics of the control rod drives. Otherwise they find the design from the safety angle satisfactory. The reactor, which is a thorium converter PWR, is fully described on pages 413-423 of this issue.

Nuclear power stations in the 500-100 MW range are being offered by International General Electric to power generating companies at firm prices and guaranteed net electrical output. The range includes 100, 200, 300, 400 and 500 MW stations and they are expected to produce electricity at a cost equal to conventional power stations in the higher fuel cost areas.

A request for $500 000 for feasibility studies on nuclear powered icebreakers has been made to Congress. It forms part of an authorization Bill for the construction of three icebreakers for the Coast Guard which is now being considered by the Committee on Merchant Marine and Fisheries.

Permission has been given by the AEC to Union Carbide for the operation of a 5 MW pool reactor in Sterling Forest, New York State.

Decontamination of the SL-1 reactor building with a view to removing and dismantling the reactor core has been approved as a better method of disposing of the reactor than waiting until natural decay of the activity. The choice was made by an AEC *ad hoc* committee (set up after the accident in January to determine the reactor's future) following consideration of proposals submitted by General Electric in late July. The committee instructed GE to proceed with their operations as soon as was practicable, emphasizing the continual need for positive administrative control during the process. They then pointed out that although it had been determined that the reactor was in a non-critical configuration it was not necessarily inherently safe. Every precaution should be taken to prevent the addition of any moderating material. They suggested that difficulties might be encountered in removing the core from the building and that an alternative scheme for disassembly of the core inside the building should be formulated. The committee supported immediate decontamination because, among other things, it might shed light on the cause of the accident, or at least lead to the elimination of some of the theories put forward. Also the demonstration of a capability to recover from an accident could be of benefit to the reactor development programme.

AEC Reorganize

CHANGES have been made in the organization of the AEC which shorten the chains of command between headquarters staff and the national laboratories and operations offices and relieve the headquarters divisions of their former supervisory and administrative functions. At the same time, the assistant general managers at headquarters (equivalent in the U.K. to assistant managing directors) are given greater responsibility. Under the old system the national laboratories were responsible to the operations offices which in turn reported to the operations divisions at headquarters, their responsibility being to the general manager. Now, the laboratories—Oak Ridge, Argonne, Livermore, Berkeley, Ames and Los Alamos—except where Project Plowshare and weapons work are concerned, are answerable on technical matters to an assistant general manager for research and development. Weapons work continues to be directed by the division of military applications, while Plowshare comes under the control of a newly created division of peaceful nuclear explosions, which reports to an assistant general manager for plans and production. Although where programmes and policies are concerned the laboratories report directly to the general manager's office, administrative responsibilities remain the concern of the operations offices. Six of these, however, now also report directly to the general manager instead of to operations divisions. They are the Chicago, Hanford, Idaho, New York, Oak Ridge and Savannah River offices. Supervisory arrangements in the case of the other AEC operations offices remain about the same as before. The function of the divisions where the laboratories and operations office are concerned now becomes largely advisory; otherwise they have similar responsibilities as before.

Correspondence

University Reactors

Sir, — The editorial article entitled "Reactor Merry - Go - Round," which appeared in your August number, contains so many misleading statements that a detailed reply would take up an excessive amount of space in your correspondence columns.

Leaving aside the many expressions of opinion, however, there is a glaring misstatement of fact which should not go unanswered. In commenting on the Imperial College reactor design you state that "Consort was very much a specialized interest of Imperial College, not easily modified to accommodate the work of other colleges and departments." Nothing could be further from the truth. Consort was conceived primarily as a neutron source reactor and its two major features of the "bare face" and the dual-purpose thermal column offer a versatility which is at least equal to that of any other neutron source reactor of comparable size and cost. The needs of other colleges in the London area were carefully considered at every stage. In no case was any experimental requirement put forward by any London college which could not have been accommodated with ease on the Consort reactor.

Your castigation of the University Committee which was set up under the chairmanship of Sir John Cockcroft is also misdirected. In their slow and hesitant way the University Authorities tried at the thirteenth hour to do the right thing in setting up this committee and in attempting to determine the relative claims and relative priorities of the various colleges. The only trouble was that they were two years too late.

Criticism should be directed not towards the University and the Cockcroft Committee but against the D.S.I.R. and the National Institute for Research in Nuclear Science. But for the intervention of these two latter bodies in the summer of 1959 in what had previously been a straightforward application from Imperial College to the University Grants Committee, the Consort reactor project could have been judged on its merits in relation to the immediate needs for teaching and research in the major engineering school of London University. Had the scheme been approved by the U.G.C. in 1959, all the other colleges in London would have benefited and the reactor would by now have been in operation and available for use by other colleges having suitably qualified staff.

By invoking the false principle that all colleges are equal, by maintaining a state of suspense for over two years, and by failing to discriminate between the immediate needs of engineering education and the hypothetical requirements of potential users of radioisotopes, the Government authorities have simply succeeded in throwing away the preparatory work of the last five years at Imperial College and in dispersing the staff of the nuclear power group who were responsible for the Consort design. Britain is virtually the only major European country in which there is now no engineering school properly equipped to teach the principles of nuclear power to the next generation of engineering graduates.

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Powder Metallurgy

Sir,—The Report of the 1961 Plansee Seminar on Powder Metallurgy on pages 388-390, *Nuclear Engineering*, September, 1961, was a thorough and useful summary of a most important event. We wish to note for your readers that the paper presented by Arbiter and Stern (discussing the preparation and properties of oxide coated uranium powder compacts prepared by hot pressing with subsequent extrusion) was submitted for the seminar by Nuclear Development of America, but that this organization became the Development Division of United Nuclear Corporation on May 31, 1961, prior to the presentation at the 1961 Plansee Seminar. Inquiries concerning this paper and related work should be addressed to:

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Critical

Ten thirty six on the night of Saturday, August 19, was an historic moment. Britain's first reactor specifically designed for electricity generation achieved criticality. It was No. 1 at Bradwell. Eight days later No. 1 at Berkeley also went critical. Following tests, loading to full power and further tests and trial runs both reactors should be in operation by early next year. This article describes the loading operations leading to criticality and the part played in these operations by the contractors, The Nuclear Power Group, the eventual owners, the CEGB, and the Ministry of Power, who are responsible to the community for ensuring the safety of commercial nuclear power plant.

UNDERSTANDABLY, the commissioning of a reactor nowadays is not quite the unpredictable process it was in the pioneer days of Windscale and Calder Hall. All the same, it can still induce a natural, healthy excitement, particularly during the stages leading to criticality. This is the first chance the engineers and physicists have of assessing the accuracy of their calculations and designs.

This stage in the commissioning of the No. 1 reactor at Bradwell and Berkeley took on added significance. As they are the first commercially owned power reactors in the U.K. it seemed to be even more vital than ever that they come up to expectation. No. 1 at Bradwell, in fact, achieved criticality with about 37·4 tons of uranium in 468 channels (each channel being loaded with only seven out of a possible eight elements), as opposed to an estimated 38 tons in 474 channels. On this basis alone it would seem that on a full loading of 235 tons in 2 564 channels ample excess reactivity would be available for power operation with an equilibrium burn-up corresponding to 3 000 MWd/t average. The loading was, in fact, over the exact critical size, a power output of 2 W being registered.

At Berkeley 740 channels were loaded, and from the neutron flux measurement at this loading the critical size calculated as being 744 channels filled with about 52·8 tons of uranium. This was 84 more channels than had been estimated.

Reactor engineering and physics apart, achievement of criticality in these reactors is also of interest from the procedural point of view. Different patterns of loading the fuel have been adopted, both on account of differences in reactor design and for providing the basis of comparative test programmes. Furthermore, being commercially owned, these reactors are the first of their kind and size to come within the scope of the Nuclear Installations (Licensing and Insurance) Act, and as such subject to inspection and general regulation by the Ministry of Power. Before loading to criticality could start, permission had to be received from the Ministry and in giving permission the Ministry also stipulated the tests to be carried out on the core when criticality had been reached.

Before going any further in describing the loading procedures, the nature of tests carried out and how the Ministry carries out its functions as a regulatory body, it is appropriate to recall some of the relevant design features of the reactors.

Each Bradwell reactor has an electrical output of 150 MW. The graphite core is approximately 40 ft dia by 25 ft 8 in high and weighs approximately 2 000 ton. There are 2 564 fuel channels each of which can be loaded with eight fuel elements, stacked one on top of each other. The elements are about 36 in long and in a fully loaded reactor there would be 20 520 of them. Weight of uranium would be about 235 ton.

At Berkeley the reactors have an individual electrical output of 137·5 MW. The core of each one is 44 ft in diameter and 24 ft high and weighs about 2 000 ton. The number of the fuel channels is 3 275, each of which can take 13 elements. These are about 19 in long and in a reactor loaded to its design power number an estimated 42 500. The uranium would weigh 232 ton.

Both stations are to be operated by the Central Electricity Generating Board and are being built for them by The Nuclear Power Group. This comprises the Nuclear Power Plant Company, who designed and started construction of the Bradwell reactor, and the A.E.I.-John Thompson Nuclear Energy Company, responsible for Berkeley.

Contracts for the stations were signed in December, 1956, and work started on site in both cases in the following month. Uranium was first loaded into No. 1 reactor at Berkeley on August 12, 1961; into Bradwell No. 1 two days later. Because of design considerations criticality was achieved in the Bradwell unit before Berkeley—the dates being August 19 and 27.

Following tests on the critical core, loading for full power operation, further tests and trial runs both reactors should be

ready for commercial operation by some time early next year. The No. 2 reactors will probably be ready six months later.

Until fuel loading starts the construction and commissioning of a nuclear power station very much follow the patterns of any other major project. Fuel loading, however, cannot be started without permission from the Ministry of Power, who have to be satisfied with the safety of the plant. Everybody working in the reactor building becomes subject to additional codes of practice and every step is taken to ensure no radiation risks are presented. Similar precautions are taken in the fuel store; and, indeed, before fuel could be taken to the stations Ministry of Power permission had to be received.

It is of interest to note here that although the reactors and their ancillary electricity generating plant are not formally handed over to the CEGB until they have been fully tested and put into operation, the CEGB are at all times responsible for safety. As licensees of the Ministry of Power they are answerable for any incidents that might arise even in the pre-operational stages. Consequently, although the contracting companies carry out the commissioning programme, the CEGB are very much in evidence as overall supervisors.

The commissioning programmes, which run into many volumes, detail not only all technical requirements of the plant but also all procedures to be adopted to ensure that nobody is exposed to risks of radiation hazard. The programmes are prepared by the Station Commissioning Committees, which are made up of CEGB staff and representatives of the contracting companies. The station superintendents act as chairmen—at Bradwell this is Mr. R. A. Peddie, at Berkeley, Mr. J. Exley. The practice of commissioning a power station by committee is new to the CEGB; hitherto it has been carried out on a more or less *ad hoc* basis!

General procedures for radiation protection and health follow the CEGB code of practice specially drawn up for use in nuclear installations. It is analogous to the Board's regulations relating to heavy engineering and electrical plant and a copy is issued to everybody working on the site. In addition, all CEGB staff—there are currently 400 at Bradwell and 283 at Berkeley—attend special nuclear energy courses.

For speed the first fuel charge at both stations is being loaded by hand. The elements are received on the pile cap from the fuel store, stripped of their polythene bags and tested for bowing and fin damage. (For reasons of space some of the initial fuel charge at Bradwell is being stored in the graphite store, whereas at Berkeley there is capacity for 500 tons in the fuel store itself. It is also of interest to note that initially the elements have been wrapped in brown paper in order to protect the outer polythene bags from the fins—a practice which we understand was later discontinued as being unnecessary.)

The elements are then handed to the loading teams. Here the procedures at the two stations differ. At Bradwell the elements are lowered singly down a standpipe by means of a simple winch from the pile cap to the core top from three different points. At Berkeley the elements are placed nine at a time into a specially designed carrier which is then lowered down a standpipe to the core top.

From the core top the elements are loaded by means of winches and grabs into the fuel channels. Loading started in the central channel and is working outwards. At Bradwell using three teams, the elements are loaded into the channels at the rate of one a minute. Loading continues round the clock, involving four shifts, each of about 36 men (Table 1). Operations are controlled from a temporary office on the pile cap

TABLE 1

Breakdown of shift for fuel loading operations on No. 1 reactor, Bradwell.

In the store	4
On the lift between store and pile cap	1
Inspection	2
Transfer of fuel to loading positions	5
Supervisor	1
Loading teams on pile cap	9
Loading teams on core top	9
Supervisor	1
Engineers/physicists	4
	36

under the charge of an engineer from the contractors—the pile cap controller—who is directly responsible to the test controller—also a contractor engineer—who has overall responsibility. (Fig. 1 shows chain of command.)

All elements are numbered and a record is kept of their position in the reactor.

During the pre-critical phase, in order to predict the eventual critical size of the core, loading was interrupted to allow the control rods to be withdrawn and neutron flux measurements made. As criticality was approached, these measurements increased in frequency. At Bradwell, for instance, the first measurement was not made until after 252 channels had been loaded; immediately prior to criticality, however, only 12 and 8 channels were being loaded between measurements.

The loading stages followed in essence the pattern outlined in Table 2. Some slight variations were made and slight supercriticality was in fact reached with 468 channels loaded as

TABLE 2

Scheduled loadings for Bradwell No. 1 and expected neutron flux measurement after each loading. In practice there were some variations, the first loading being, for instance, of 252 channels instead of 240 and some of the later loadings being 16 instead of dozens or eights. Supercriticality was in the end reached with 468 channels.

No. each loading	Total loaded	Expected neutron count
240	240	65·3 n/cm²/sec
90	330	142
60	390	299
24	414	449
12	426	582
12	438	841
8	546	1 163
8	554	1 862
8	562	4 732
12	574	

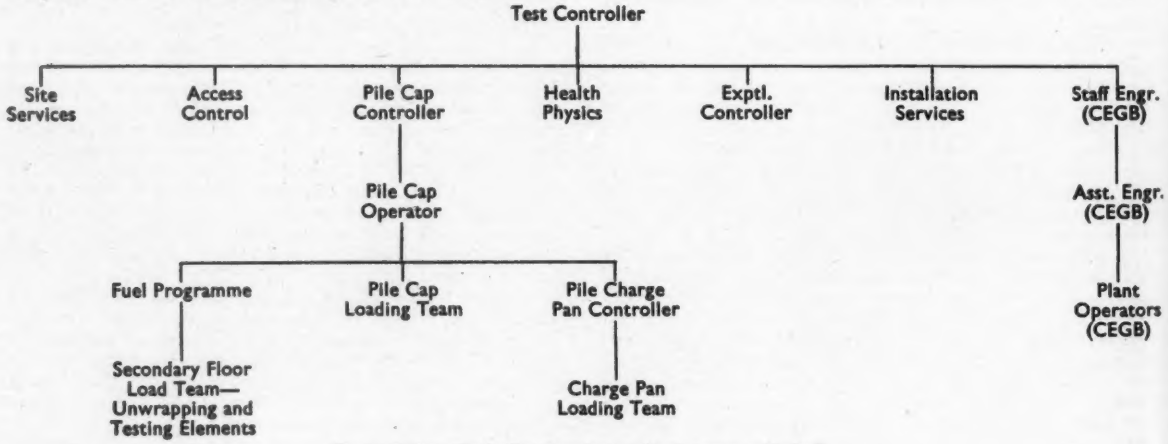

Fig. 1.—Staff tree for loading operations on No. 1 reactor at Bradwell.

against an estimated 472. Power level recorded was 2·5 W(th). Fig. 2 shows general patterns of loading of the critical core together with instrumentation.

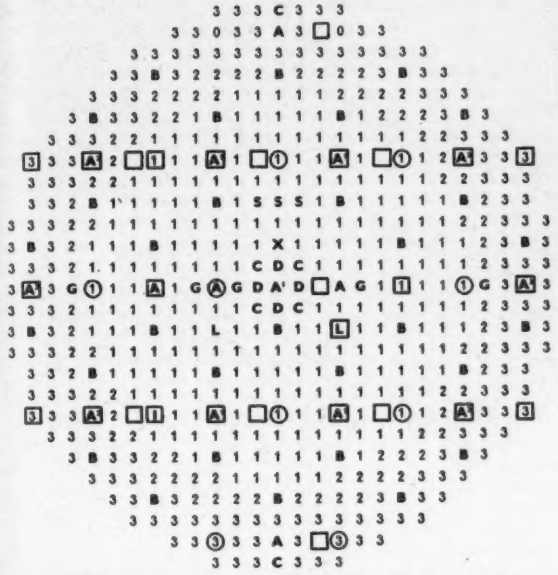

Fig. 2.—Bradwell No. 1 reactor core plan showing instrumentation and main stages in loading to criticality. Key: A absorber channels; B bulk control rods; C 12EB 40 counters; D DC amplifiers; G graphite sampling channels; L low level log; S shutdown amplifier; X safety rod; □ flux scanning; O loading points.

At Berkeley the first flux measurement was made after only 62 channels had been loaded. This was to check the six ion chambers and four boron counters located in the top of the core. The next measurement was made after 393 channels had been loaded; subsequent measurements taking place as indicated in Table 3 and a graph for determining critical load prepared as shown in Fig. 5.

TABLE 3

Loading of Berkeley No. 1 reactor to criticality and the neutron counts for predicting critical size.

Channels loaded	Neutron count rate	Channels loaded/neutron count ratio
62		
298		
396		
448		
496	66 n/cm²s	7·6
584	119	4·89
639	198	3·23
672	310	2·16
696	505	1·38
712	790	0·90
728	1 660	0·44
740	6 200	0·11
744		

Fig. 4.—The nine element carrier being lowered to the core top at Berkeley.

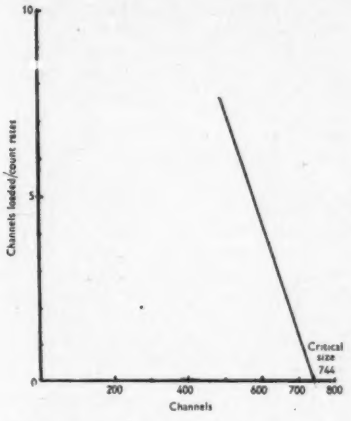

Fig. 5.—Graph to predict critical size at Berkeley extrapolated from channels loaded and ratio of channels loaded to neutron count rate.

A considerable amount of time during the pre-critical phase was taken up by the safety procedures adopted for carrying out the neutron flux measurements. Movement of personnel to and from the reactor vessel is controlled by the CEGB "permit-to-work" system. Under this, nobody can enter the vessel unless there is a written permit signed by the shift charge engineer in force.

While men are inside the reactor pressure vessel the bulk

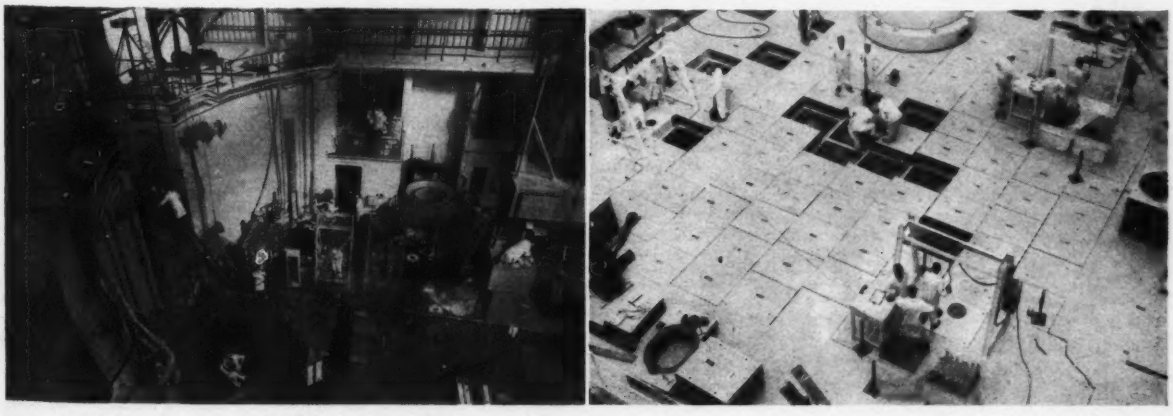

Fig. 3.—The pile caps of No. 1 reactors at Berkeley (left) and Bradwell during loading operations.

control rods are locked down in position so as to prevent accidental withdrawal. When a measurement is to be made the men leave the vessel—access incidentally is via a manhole in one of the gas ducts—and the PTW, as the permit is called, cancelled. The test controller, experimental controller and pile cap controller are informed and the control rods energized, withdrawn and the count made. To speed up the experiment, the rods were removed 100 times faster than their normal speed.

After the count the rods were dropped back in position and the pressure vessel monitored by a health physicist. Background radiation was invariably found to be less than that on the pile cap or in the control room. Loading could not restart, however, without a new PTW being issued.

In emergencies the reactor safety rods could, of course, be used. These, like the bulk rods, were set at Bradwell to trip at a power level of 10 W equivalent to a neutron flux of $1{\cdot}7 \times 10^6$ n/cm²s.

Although there is no danger in entering the pressure vessel during loading, at Berkeley, due to misunderstanding by some of the loading teams, it was found necessary to provide the men with additional assurances of the safety. The CEGB's chief medical officer of health gave a lecture, and a letter was sent to everybody concerned outlining the health physics involved.

Everybody working in the reactor areas has, of course, to wear a film badge, while men working inside the pressure vessel were subjected to a vigorous medical examination and blood counts. Strict measures were also followed in observing "clean conditions," the loading teams being required to make a complete change of clothing, including underwear, before entering the pressure vessel or working on the standpipes from the pile cap.

Once criticality was reached the cores in both reactors were subject to a series of tests and experiments. These differed with the two reactors not only on account of design differences but because there is still very much to be learnt about commissioning power reactors.

The Criticality Tests

One of the main differences between the Bradwell and Berkeley reactors was that the former was taken to criticality with each channel loaded with one element less than its full complement. After some tests and measurements had been made the eighth element was added and further tests carried out. In this way it was possible to obtain both comparative axial and radial laplacians instead of just comparative radial measurements as in the core of Berkeley No. 1, where channels were filled with their full complement right from the start.

Among the measurements and tests carried out on the Bradwell core were, with the first loading: reactivity as a function of loaded radius; air pressure coefficient of reactivity; radial and axial laplacians; observance of the effect of inserting a grab head into a channel and then removing fuel; the effect of inserting an unirradiated antimony/beryllium source; and observing the effect of loading a channel with an absorber and measuring the fine flux.

After the channels had been loaded to the full the built-in reactivity was checked, radial and axial laplacians calculated and the reactivity held by one control rod measured. Results of these tests, unfortunately, are not yet generally available.

At Berkeley tests included the measurement of variation in reactivity of the core with channels loaded above the critical size; measurement of the radial and axial laplacians; observation of the effect of lowering two central control rods and measuring the flux distribution with these rods partially inserted; comparison of the effects of using stainless steel and zirconium brackets on the fuel elements (zirconium brackets are to be used on elements in the outer zone of the core); and the effect of inserting the charge machine grab head into a channel.

These particular tests, and the loading of the Bradwell channels with only seven elements, were in fact specified by the Ministry of Power in their permit for loading to criticality.

It is opportune here to examine the general functions and powers of the Ministry as a nuclear regulatory body.

Under the Nuclear Installations (Licensing and Insurance) Act, passed in 1959, all non-AEA, non-Government owned reactors and other nuclear installations have to be licensed by, in England and Wales, the Ministry of Power; in Scotland, the Secretary of State; and in Northern Ireland, the Minister for Commerce. The function of these bodies is to ensure that the licensees build and operate their reactors without risk of hazard to the community. They lay down the safety requirements for each plant and can at any time visit it, require tests to be made, create new conditions to the licence and even revoke it.

To carry out these functions in England and Wales the Ministry of Power has established an Inspectorate of Nuclear Installations. This is made up of scientists, engineers and physicists drawn from the AEA, industry and other sources, under the direction of Major-General S. W. Joslin. There is a total of 34 people at present in the Inspectorate. Most of them are engaged on plant optimization and safety assessment, as well as on actual inspection of sites.

Permits Required in Stages

Under the terms of the currently held CEGB station licences, which, incidentally, cover both reactors to each station, the licensee is obliged to seek the Ministry's permission before proceeding with certain stages in both the commissioning and operating programmes. There is no set formula for deciding these stages and the Inspectorate is, in fact, basing its requirements on an assessment of each stage as it comes. At Bradwell and Berkeley permits have been required for taking fuel to the station sites, loading the No. 1 reactors to criticality and loading for full power operation. Further permits are expected to be required for testing at various stages and operation at different power levels. Permits can only be issued by the Chief Inspector, who is directly responsible to the Minister.

As well as giving approval to the start of any particular stage, a permit can specify the method to be adopted for carrying out that stage and the tests required to be carried out on completion of that stage, as for example in the case of the permits to load to criticality for No. 1 reactors at Bradwell and Berkeley. Good results do not, however, lead automatically to the granting of a permit for the next stage.

Naturally enough, there is a considerable amount of co-operation between the Inspectorate and CEGB. The Inspectorate will, for instance, discuss with the Board tests it considers vital and whether or not they should be carried out in the presence of Ministry staff. Advice will be given on tests the Inspectorate think might be of use to the Board but in which they themselves are not initially interested, and they will specify plant and equipment that they consider not to be of their concern at all. The Inspectorate is also always in close touch with the AEA, particularly on safety assessment matters.

Overall, the Inspectorate sets out to be anticipatory in its actions rather than retrospective. Even so, if there is need for restraint, the Inspectorate will not hesitate in using its powers, as in fact it has already demonstrated.

Permission to load the No. 1 reactor at Berkeley was given a week after application had been made; at Bradwell the process took only two days. Apparently the Ministry would not give permission for the Berkeley loading until supplied with certain additional information, hence the delay.

It was a small matter, and there were no bad feelings or argument about it. It did show, however, that the Inspectorate intends to be independent and unafraid of doing its job.

Naturally enough, the incident also enhanced the harmless sense of rivalry that exists between the construction and commissioning staffs at both stations. Bradwell, which started loading two days after Berkeley, did, in fact, reach criticality first, mainly because of design reasons. The difference in achievement of criticality, however, would have been much smaller if there had been no delay over the Berkeley licence and possibly there would have been a chance of them beating Bradwell.

Although Bradwell is, so to speak, currently in the lead, this is no guarantee that Bradwell No. 1 reactor will be the first to produce heat for sending electricity into the grid. However, from what has happened during loading to criticality, and from the obvious aim of The Nuclear Power Group, the CEGB and the Ministry of Power to work as closely as possible, what race there is will not be won at the expense of any desirable caution.

Inhibition of Stress-corrosion Cracking of Austenitic Stainless Steel

The effects of a number of inhibitors on the stress-corrosion behaviour of an 18/9 Cr Ni, Nb stabilized, austenitic stainless steel have been investigated in tests closely simulating severe conditions likely to be met in practice.

In chloride environments tri-sodium phosphate, Na_3PO_4, present in a ratio of not less than 1 : 2, tri-sodium phosphate: sodium chloride, has been shown to prevent, or at least greatly delay, stress-corrosion cracking. Its use in conjunction with other currently accepted methods of protection would significantly reduce the risk of attack under actual operating conditions.

by P. P. SNOWDEN, A.Met.
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AUSTENITIC stainless steel, like most other alloys, can suffer stress-corrosion cracking under certain conditions of stress and corrosive environment. The appropriate circumstances are sometimes encountered in industrial plant and can result in failure of the particular item of equipment involved. There is now considerable knowledge of the subject and much can be done, by way of design and operation of plant, to prevent this form of attack, but there are still occasions when the use of an inhibitor could be beneficial.

In practice, the chief cause of failure is the presence of chloride, even in very small amounts, in hot solutions in contact with the steel. Laboratory tests have shown that even very dilute solutions can cause some attack, but where conditions allow concentration of chloride from an initially very dilute solution, then attack can be very rapid. Such conditions could occur in a heat-exchanger of the type which might be used between primary and secondary circuits of a reactor operating with a natural water of low chloride content. In such an application, where high cost is involved and absolute integrity required, the use of an inhibitor to supplement other preventive measures would be a useful precaution.

To determine the effects of a number of possible inhibiting substances on the behaviour of a typical austenitic stainless steel, an extensive series of tests has been carried out under severe laboratory conditions.

Material

The steel used in this work was an 18/9 Cr Ni steel stabilized with niobium, a typical analysis being, C 0·07, Si 0·60, Mn 0·78, Cr 17·72, Ni 8·75, Nb 0·80, but it has previously been shown[2] that behaviour of all the "18/8" type steels is similar and the present results would be expected to apply equally well to any steel of this type.

Test Procedure

Stress-corrosion cracking of austenitic stainless steels has received a considerable amount of attention in recent years, as even a brief survey of the literature shows, but much of the work has either been concerned with theoretical aspects of the mechanism of the attack or has dealt with relatively simple practical cases. Where laboratory tests have been carried out, they have largely employed environments of little direct practical importance, such as boiling 42% magnesium chloride solution. While tests of this type have led to an understanding of the mechanism and so enabled recommendations to be made to reduce the incidence of attack, they do not permit the evaluation of substances likely to be useful as inhibitors under practical conditions. For work of this type it is necessary to reproduce, as nearly as possible, the more severe environments which may be met under operating conditions. The behaviour of austenitic steels in tests under such conditions has been described elsewhere, along with a detailed description of the apparatus used[3, 4]. Tests were carried out with pressurized solutions at 250–300°C; with contaminated surfaces exposed to condensing (dewpoint) or slightly superheated steam, at 1 500 psi; and under concentrating conditions, at 330°C, 1 500 psi, using initially very dilute solutions. Both chloride and caustic alkali environments were studied and rapid stress-corrosion cracking was found under all conditions.

In the present work, the same apparatus and test procedures have been used, except that varying amounts of different chemicals thought to be of interest, have been added to the sodium chloride or sodium hydroxide used in the test. As chloride environments are responsible for most service failures, the greater part of the present work was with sodium chloride, but sodium hydroxide can also cause rapid stress-corrosion cracking and this was used in some of the tests. The following are the chemicals selected for study: sodium phosphates NaH_2PO_4, Na_2HPO_4 and Na_3PO_4; sodium nitrate $NaNO_3$; sodium nitrite $NaNO_2$; sodium chromate Na_2CrO_4; sodium sulphate Na_2SO_4; sodium silicate Na_2SiO_3; sodium bi-sulphate, $NaHSO_4$ and potassium phosphate K_2HPO_4.

Chloride Environments

Under the very severe conditions of the dewpoint test in high pressure steam, in which, with plain chloride contamination, complete fracture occurs within a few minutes of condensation starting, the presence of equal amounts of Na_3PO_4 to NaCl completely prevented attack. With lower ratios some cracking was found, although complete fracture was prevented. In high temperature solutions, in which with plain chloride solutions rapid fracture again occurs, a ratio of 1 : 3 Na_3PO_4/NaCl was sufficient to prevent attack, while still lower ratios of 1 : 4 and

TABLE I. Tests in steam at 300°C or 330°C, 1 500 psi pressure including 100 psi air. Specimen stressed to 13 t/in²

Condensing (Dewpoint) steam at 300°C Ratio of NaCl/Inhibitor in saturated solution used to coat specimen	Result of 1 cycle test	Observations
NaCl/Na_3PO_4 1/1	Not failed	No cracking
NaCl/Na_3PO_4 2/1	Not failed	Several large cracks
NaCl/Na_2HPO_4 1/1	Not failed	No cracking
NaCl/K_2HPO_4 1/1	Not failed	Several large cracks
NaCl/NaH_2PO_4 1/1	Not failed	Many medium deep cracks
NaCl/NaH_2PO_4 2/1	Failed	Many medium deep cracks besides fracture
NaCl/$NaNO_2$ 1/1	Not failed	A few large and several small cracks
NaCl/$NaNO_2$ 2/1	Failed	No cracking other than fracture
NaCl/$NaNO_3$ 1/1	Failed (two tests)	Many deep cracks
NaCl/Na_2SO_4 1/1	Failed (two tests)	Many deep cracks
NaCl/Na_2CrO_4 1/1	Failed	Many deep cracks
NaCl/Na_2SiO_3 1/1	Failed	Many deep cracks
Typical result with plain NaCl	Failed	Many deep cracks

Superheated steam at 330°C Ratio of NaCl/Inhibitor in saturated solution used to coat specimen	Duration of test in hours	Observations
NaCl/Na_3PO_4 1/1	1 436 NF*	No cracking
NaCl/Na_2HPO_4 1/1	3 506 NF	Two large and many small cracks
NaCl/$NaNO_2$ 1/1	1 510 NF	Severe cracking, almost complete penetration at one point
Typical result with plain chloride	96	Severe cracking

* Not failed.

TABLE 2. Tests in inhibited sodium chloride solutions at 250°C under nitrogen at 1 000 psi plus air at 200 psi. Specimen stressed to 10 t/in²

Ratio of NaCl/Inhibitor in 3% NaCl solution	Duration of tests in hours	Observations
NaCl/Na_3PO_4 1/1	86 NF*	No cracking
NaCl/Na_3PO_4 2/1	697 NF	Very slight cracking
NaCl/Na_2HPO_4 1/1	117	Severe cracking
NaCl/$NaHPO_4$ 1/1	1·3, 5·0	Some cracking
NaCl/$NaNO_2$ 1/1	671 NF	No cracking
NaCl/$NaNO_2$ 4/1	29 NF	Several deep cracks
NaCl/Na_2SO_4 1/1	45 NF, 113 NF	No cracking
NaCl/Na_2SO_4 3/1	25 NF	Severe cracking
Typical result with plain 3% NaCl	1·0	Severe cracking

1 : 6 gave considerable protection but did not completely prevent attack. Under concentrating conditions (in a thermal loop) closely representative of practice, very good protection was provided by a ratio of 1 : 2 Na_3PO_4 to NaCl. Other inhibitors gave varying degrees of protection, depending on the type of test. Sodium nitrite gave good results in steam and high temperature solutions; the other phosphates NaH_2PO_4, Na_2HPO_4, K_2HPO_4, and sodium sulphate gave some protection under certain conditions, but sodium nitrate and sodium chromate, which are oxidizing agents, appeared to accelerate attack.

The best overall protection was undoubtedly afforded by tri-sodium phosphate, Na_3PO_4, when present in a ratio of not less than 1 : 2. Changes in the conditions such as reduction of stress, lack of adequate oxygen (required for chloride stress corrosion) or increased superheat, may reduce the severity of the test with plain chloride[3, 4]. A selection of results illustrating the protection afforded by the various inhibitors under chloride environments is given in Tables 1, 2 and 3.

Caustic Alkali Environments

From the practical aspect, the problem with caustic alkalis appears to be much less acute and, although occasional failures are encountered, the incidence of cracking due to this cause is much less than with chloride. It was, however, considered worth while to examine some of the inhibitors with caustic soda and tests were carried out in steam and high temperature solutions. Generally the extent of inhibition was less than with chloride, but limited protection was found with tri-sodium phosphate in high temperature solutions, this being critically related to the $NaOH/Na_3PO_4$ ratio. Where chemical reaction was possible, as with sodium bisulphate, some of the protection was probably due to reduction of the effective NaOH concentration.

The slight alkalinity conferred by tri-sodium phosphate could be used to protect mild steel components in composite mild steel—stainless steel equipment, such as may be found in some nuclear power plant, where it is at present the practice to add sodium, potassium or lithium hydroxide to maintain an alkaline pH. To investigate the possibility of trouble from phosphate-induced alkalinity, in terms of stress-corrosion of the stainless steel, tests with plain Na_3PO_4 solutions were carried out in the thermal loop and in the high temperature solution apparatus. The tri-sodium phosphate was dissolved in demineralized water and the pH of the solution measured at room temperature. In the thermal loop a solution of room temperature pH 10·9 caused no attack in 2 000 h. In the high temperature solution apparatus, a solution of room temperature pH 10·2 gave no attack in 500 h, but solutions of room temperature pH 11·0 and 11·2 caused failure in 15 and 32 h, respectively. A selection of the results with caustic environments is given in Tables 4 and 5.

Discussion of Results

The mechanism of the inhibiting process was not directly investigated. Consideration of the results suggests, however, that it is an electrochemical rather than a simple blanketing action (as in the use of sodium sulphate in ordinary boiler practice for the prevention of caustic embrittlement) because protection was found in high temperature solutions from which no precipitation should occur. If this is the case, it is very important to remember that anodic inhibitors, i.e., those which stifle the corrosion reactions at the anode, can, if added in insufficient quantity for complete inhibition, localize and actually intensify the attack. Cathodic inhibitors, on the other hand, stifle the corrosion reaction at the cathode, and whilst they will not intensify attack regardless of the amount present, it is much more difficult to completely prevent attack by their use. Of the inhibitors used in the present work with chloride, only two have actually accelerated attack, sodium nitrate and sodium chromate, and here it is not certain whether this was due to their being anodic inhibitors, and present in less than the required minimum amount, or to the fact that both are oxidizing agents, which could also lead to acceleration.

With the other salts, where cracking had occurred, it was usually of the branched transcrystalline type typical of stress-corrosion in chloride environments (see Fig. 1). Some unfailed samples contained a number of very small cracks, while others contained a few isolated, very deep cracks, often straight and unbranched. Where there had been a considerable increase in life, both these types of cracking were taken to indicate almost complete protection. This could be attributed to cathodic inhibition in the case of the small cracks, while in the second case the localization of the attack along with an increase of life suggests that the inhibitor was operating at both the anodic and cathodic areas. Under the relatively mild conditions of the superheated steam test, isolated cracks were frequently associated with deep rounded pits. Fig. 2 illustrates this type of attack. Of the two more successful inhibitors, tri-sodium phosphate and sodium nitrite, the latter sometimes localized attack and so appears to be less safe than the tri-sodium phosphate, which gave the best overall protection with no isolated deep cracks.

With sodium hydroxide the extent of the inhibition was generally less than was found with chloride; only tri-sodium phosphate over a limited range, gave any appreciable protection. The effect was in fact found to be so critically related to the actual $NaOH/Na_3PO_4$ ratio that it is unlikely to have any practical advantage. This protective effect was first noticed with the additions of the acid phosphate, NaH_2PO_4, to NaOH in amounts which, assuming reaction with the NaOH, resulted in a ratio of 17/7 NaOH to Na_3PO_4 (Table 5). Subsequently it was found that provided this ratio was maintained, the addition of Na_3PO_4 to sodium hydroxide gave protection only if the NaOH concentration did not exceed 17%. Deviation from this ratio, or raising the sodium hydroxide concentration to 20% whilst maintaining this ratio, led to rapid cracking.

One other interesting feature of this work with caustic soda has been the effect of inhibitors on the nature of the cracking. In high pressure steam the attack on plain NaOH coated specimens is almost entirely transcrystalline, while in high temperature solutions of NaOH (of the strengths considered here) attack is entirely intercrystalline. In the present work, in many of the tests in both steam and solutions, the reverse has been found. Sodium nitrite and sodium sulphate led to intercrystalline cracking and general intercrystalline corrosion in the high pressure steam tests, while tri-sodium phosphate caused a change to

TABLE 3. Tests under concentrating conditions in a thermal loop at 1 500 psi pressure including 200 psi air

Initial solution	Duration of test in hours at various temperatures 330%C	350%C	370%C
100 ppm NaCl/100 ppm Na_3PO_4	170, 216	350 NF, 1 000 NF	
100 ppm NaCl/50 ppm Na_3PO_4	350, 500	177, 600 NF	
50 ppm NaCl/25 ppm Na_3PO_4	126, 180	716	
100 ppm NaCl/100 ppm Na_2HPO_4	360	60, 108	480, 650
100 ppm NaCl/100 ppm NaH_2PO_4	90	1 000 NF	
100 ppm NaCl/100 ppm $NaNO_2$	60, 66, 100	45	
100 ppm NaCl/100 ppm Na_2SO_4	176, 350	464	
Typical results with plain NaCl. 100 ppm NaCl	20–24		18

* Not failed

TABLE 4. Tests in steam at 330°C, 1 500 psi pressure including 100 psi air. Specimens stressed to 13 t/in²

Ratio of NaOH/Inhibitor in 30% NaOH solution used to coat specimen	Duration of test in hours	Observations
NaOH/Na_2HPO_4 2/1	5	Many small cracks besides fracture
NaOH/Na_2HPO_4 1/2	460 NF*	A number of very small cracks
NaOH/NaH_2PO_4 1/1	117 NF	No cracking
NaOH/$NaNO_3$	966 NF	Several large cracks, general intercrystalline attack
NaOH/$NaNO_3$	313	Many small cracks besides fracture
NaOH/Na_2SO_4	1 112	Intercrystalline fracture, general intercrystalline attack
NaOH/NaHSO₄	1 958 NF	No cracking
Typical result with plain NaOH	110	Severe transcrystalline cracking

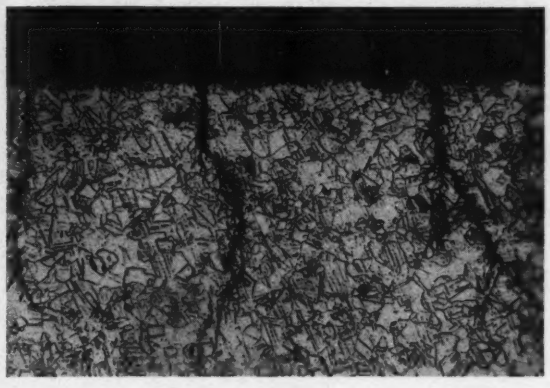

Fig. 1.—Photomicrograph of typical chloride-induced stress corrosion cracking.

transcrystalline cracking in high temperature solutions.

The failures experienced with plain phosphate solutions of high pH values suggest that the use of tri-sodium phosphate alone to raise pH may not be as safe as was thought. In the tests with solutions of pH 11·0 and 11·2 at room temperature, extensive slight intercrystalline attack and one or two deep intercrystalline cracks were present. The fracture was also intercrystalline and this is typical of attack by high temperature caustic soda solutions. Furthermore, although no intercrystalline attack was found with phosphate inhibited chloride solutions, where the specimen did not fail, a number of specimens which did fail were found to have intercrystalline cracking on both the inside and outside surfaces. This was attributed to solution drying out after fracture and breakdown of the inhibitor.

Practical Implementation of the Results

Tri-sodium phosphate has been shown to reduce greatly the possibility of stress-corrosion cracking of austenitic stainless steel under very severe conditions closely resembling those likely to be met in practice, but it is not suggested that its addition should be solely relied upon to prevent attack under actual service conditions. Rather it should be regarded as a supplementary means of protection to be used along with the currently accepted methods of reducing attack. These have been mentioned by various authors and have been discussed in detail by Edeleanu[5], but briefly it may be said that removal of residual stresses by suitable heat treatment, where this can be safely accomplished, and de-aeration of plant and feed waters, greatly reduce the risk of attack in chloride environments. In addition, anything which reduces the severity of the environment; keeping temperature to a minimum, reducing applied stresses and keeping chloride concentration as low as possible, is beneficial. In this latter context, much can be done at the design stage to prevent concentration from initially dilute solutions by avoidance of crevices or local hot spots.

Unfortunately all these recommendations cannot always be complied with and it is then that the use of an inhibitor may be especially beneficial. A typical example might be where concentration from an initially quite dilute solution, say 100 ppm chloride, could occur in crevices between the tubes and end plate of a heat exchanger. Under such conditions, the addition of 100 or even 50 ppm Na_3PO_4 to the solution, along with observation of any or all of the previous recommendations should prevent, or at least greatly delay, stress-corrosion cracking.

The problem of " hide-out " of phosphate often encountered in the operation of ordinary boilers can be prevented by using potassium salts[6]. A limited number of tests, which have been carried out in the present work, have indicated that the behaviour of K_3PO_4 is similar to that of Na_3PO_4, so that where it is proposed to employ an inhibitor and conditions may be such as to lead to " hiding-out," the use of the tri-potassium rather than the tri-sodium phosphate may be preferable.

ACKNOWLEDGMENT

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Fig. 2.—Isolated cracking and pitting found with $NaCl/NaNO_2$ coated specimen exposed to superheated steam.

TABLE 5. Tests in solutions at 300°C under nitrogen at 1 000 psi plus air at 200 psi. Specimens stressed to 10 t/in²

% NaOH and Inhibitor	Duration of test in hours	Observations
20% NaOH–5% NaH_2PO_4	67	Many transcrystalline cracks, a few intercrystalline cracks
17% NaOH–7% Na_3PO_4	217, 679 NF	No cracking, general corrosion in failed specimen
10% NaOH–4% Na_3PO_4	363	Transcrystalline fracture, no other cracking
5% NaOH–5% Na_3PO_4	3·0	Transcrystalline fracture, no other cracking
20% NaOH–8% Na_3PO_4	2·0, 2·6	Transcrystalline fracture, a few transcrystalline cracks
5% NaOH–5% $NaNO_3$	24·7	Failure due to general corrosion, no cracking
5% NaOH–5% Na_2SO_4	32, 48	Transcrystalline cracking
5% NaOH–5% NaHSO₄	138	Mixed inter and trancrystalline fracture, no other cracking
5% NaOH–5% Na_2CrO_4	9	Intercrystalline fracture, no other cracking
Typical results with plain NaOH solutions 20%	1·8	Severe intercrystalline cracking
5%	15·1	Severe intercrystalline cracking

* Not failed

TRANSLATIONS

Bradwell et Berkeley critiques.

Le chargement des deux premiers réacteurs des deux premières centrales nucléaires industrielles britanniques—Bradwell et Berkeley—a été entrepris en août. A la suite d'essais étendus et de périodes de marche expérimentales les réacteurs seront accouplés aux générateurs d'électricité et l'alimentation du secteur commencera au début de l'année prochaine. Les deux réacteurs ont été critiques conformément aux hypothèses des bureaux d'études. Bradwell No. 1 a été critique six jours après le début du chargement qui a nécessité 37,4 tonnes d'uranium. Les chiffres correspondants de Berkeley sont quinze jours et 52,8 tonnes.

Inhibition de la fissuration par la corrosion due aux contraintes dans les aciers inoxydables austénitiques

Les effets de certains inhibiteurs sur le comportement d'un acier inoxydable austénitique 18/9 Cr Ni stabilisé, lors de la corrosion par les contraintes, ont été étudiés lors d'essais reproduisant fidèlement les conditions sévères qui peuvent exister en pratique.

Dans un milieu riche en chlorures, où le tri-phosphate de sodium Na_3PO_4, est présent dans un rapport qui n'est pas inférieur à 1/2, on a pu montrer que le chlorure de sodium peut empêcher la corrosion sous contraintes, ou tout au moins retarder considérablement la fissuration qui s'ensuit. Son utilisation, en même temps que d'autres méthodes de protection qui ont fait leurs preuves réduirait considérablement le risque d'attaque dans les conditions actuelles d'utilisation.

Indian Point

La centrale d'Indian Point, sur la rivière Hudson, à environ 40 km au nord de New York, est actuellement construite par la Consolidated Edison Company. Le réacteur à eau sous pression, a été conçu par Babcock et Wilcox, est un convertisseur de thorium. La puissance électrique brute de la centrale sera de 275 MW dont 112 MW produits dans un surchauffeur à mazout.

Orifices à bords renforcés—2ème partie

Dans la première partie (Nuclear Engineering, août 1961, p. 334), il a été donné une description des méthodes pouvant être utilisées pour calculer les contraintes dans le voisinage des orifices à bords renforcés dans un réservoir soumis à la pression. Des méthodes permettant de tenir compte de poussées sur les conduites soudées à l'emplacement des ouvertures ont également été données. Ces méthodes sont maintenant développées, de telle sorte que les contraintes dues à un moment fléchissant s'exerçant sur une conduite puissent être estimées dans le voisinage de celle-ci. Les résultats obtenus par les deux méthodes sont comparés avec ceux que l'on obtient lorsqu'un moment fléchissant est appliqué à un élément rigidement fixé à une plaque plate infinie.

L'extraction de l'iode radioactif des effluents gazeux.

Dans un article antérieur par Geldart les problèmes que pose l'extraction de l'iode radioactif des circuits gazeux dans les réacteurs ont été discutés et l'efficacité de différents adsorbants a été évaluée. Dans le présent article, les problèmes relatifs aux réacteurs civils gaz-graphite sont traités de façon encore plus détaillée. Il en est de même d'expériences au cours desquelles des essais ont été effectués avec du carbone, du cuivre, du bronze et de l'argent comme adsorbants.

Kritisch—Bradwell und Berkeley.

Mit dem Aufladen der ersten Reaktoren in den beiden ersten britischen kommerziellen Atomkraft–Werken—Bradwell und Berkeley—wurde im Laufe des August begonnen. Nach ausgedehnten Versuchen und Probebetriebs–Perioden werden die Reaktoren mit der Elektrizitäts-Erzeugungsanlage verbunden werden, und Strom wird Anfang des nächsten Jahres in das Netz geschickt werden. Beide Reaktoren haben den kritischen Zustand erreicht, und es scheint, dass sich die Erwartungen der Konstruktöre erfüllt haben. Bradwell No. 1 erreichte den kritischen Zustand sechs Tage nach Beginn des Aufladens mit 37,4 t. Uranium, und Berkeley No. 1 wurde kritisch fünfzehn Tage nach Beginn des Aufladens mit 52·8 t.

Die Inhibierung von Rissen durch Rosten unter Beanspruchungen bei austenitischem rostfreien Stahl.

Die Wirksamkeit einer Anzahl von Inhibierungsstoffen in Bezug auf das Verhalten von 18/9 CrNi, Nb stabilisiertem, austenitischen rostfreien Stahl gegenüber Rosten unter Beanspruchung ist durch Versuche erforscht worden, bei denen man die schweren Bedingungen, die in der Praxis auftreten, aufs genaueste nachgeahmt hat. In einer Chloride enthaltenden Atmosphere hat es sich gezeigt, dass Tri-Natriumphosphat, Na_3PO_4, vorhanden im Verhältnis von mindestens 1:2 als Tri-Natriumphosphat: Natriumchlorid, verhindern oder es zum mindesten stark verzögern kann, dass Risse durch Rosten unter Beanspruchung auftreten. Seine Verwendung in Verbindung mit anderen allgemeinen akzeptierten Methoden des Schutzes würden das Risiko des Angriffs unter den in der Wirklichkeit auftretenden Betriebsbedingungen ganz bedeutend reduzieren.

Indian Point.

Das Kraftwerk Indian Point am Hudson, 40 km nördlich von New York wird zur zeit von der Consolidated Edison Company gebaut. Der Druckwasser-Reaktor, nach dem Entwurf von Babcock and Wilcox, ist ein Thorium Wandler. Die elektrische Gesamtleistung wird 275 MW sein, von der 112 MW durch oelgefeuerte Ueberhitzung erzeugt wird.

Rand-verstärkte Oeffnungen—Teil II.

Im Teil I (Nuclear Engineering, Aug. 1961, Seite 334) wurde eine Beschreibung der Methoden gebracht, mit denen man die Beanspruchungen in der Nachbarschaft von randverstärkten Oeffnungen in einem Behälter berechnen kann, der Drücken unterworfen ist. Es wurden weiter Methoden gezeigt, mit denen man Drücke am Ende einer Abzweigung behandeln kann. Diese Methoden sind nun weiter ausgebaut worden, so dass man auch die Beanspruchungen schätzungsweise bestimmen kann, die um eine Abzweigung herum auftreten, die einem Biegungsmoment ausgesetzt ist.

Die Ausscheidung von radioaktivem Jod aus den Abgasen.

In einem früheren Aufsatz, von Geldart, sind die Probleme besprochen worden, die sich bei der Ausscheidung von Jod aus den Gasströmen. Die durch den Reaktor gehen, ergeben, und der Wert der verschiedenen Stoffe zur Adsorbierung ist untersucht worden. In dem vorliegendem Aufsatz werden die besonderen Probleme behandelt, die bei Reaktoren vom Typ des Calder Reaktors auftreten.

Crítico—Bradwell y Berkeley.

La carga de los primeros reactores en las dos primeras centrales nucleares comerciales de la Gran Bretaña—Bradwell y Berkeley—comenzó durante el mes de Agosto. Después de largas pruebas y períodos de marcha experimental los reactores van a conectarse a la instalación generadora de energía y la red empezará a funcionar a principios del año próximo. Ambos reactores han alcanzado criticidad y parece ser están dando los resultados que esperaban sus creadores. El reactor Bradwell No. 1 se hizo crítico seis días después de empezar la carga con 37,4 toneladas de uranio y el Berkeley No. 1 quince días después de comenzar la carga con 52,8 toneladas.

Inhibición del agrietamiento por corrosión con esfuerzo del acero inoxidable austenítico.

Durante ensayos en los que se han simulado muy de cerca las condiciones durísimas que más probablemente han de prevalecer en la práctica se han estudiado los efectos de diversos inhibidores sobre el comportamiento ante la corrosión con esfuerzo de un acero inoxidable austenítico de 18/9 Cr Ni, estabilizado al Nb.

En ambientes de cloruro fosfato trisódico, Na_3PO_4, presente en una proporción de no menos de 1 : 2, fosfato trisódico: cloruro de sodio, ha demostrado poder evitar, o al menos retardar mucho, el agrietamiento por corrosión con esfuerzo. Su empleo en combinación con otros medios corrientes de protección reduciría de manera interesante el riesgo de ataque en condiciones normales de funcionamiento.

Indian Point.

La central eléctrica de Indian Point que se está construyendo sobre el río Hudson, a unos 38 kilómetros al norte de la ciudad de Nueva York, es obra de la Consolidated Edison Company. El reactor con agua a presión, una creación de Babcock and Wilcox, es del tipo convertidor de torio. La producción eléctrica bruta es de 275 MW de los cuales 112 MW se producen con supercalor alimentado por aceite.

Aberturas reforzadas con pestaña—Parte 2a.

La primera parte (Nuclear Engineering, Agosto 1961, pág. 334) contenía una descripción de los métodos empleados para calcular los esfuerzos en las cercanías de aberturas reforzadas con pestaña en un recipiente sometido a presión. También se explicaban métodos para el tratamiento do empujes de extremo en un tubo unido a un recipiente bajo presión. En este parte se amplian estos métodos de forma que pueden calcularse los esfuerzos alrededor de un tubo sujeto a un momento de flexión. También se comparan los resultados de estos dos métodos con los que se obtienen cuando se aplica un momento de flexión a un aditamente rigidamente conectado a una plancha plana infinita. La notación aparece en la Parte 1a.

Eliminación del yodo radiactivo de la corriente efluente gaseosa.

En un trabajo anterior de Geldart se han estudiado los diversos problemas que surgen en la eliminación del yodo presente en las corrientes gaseosas del circuito del reactor y se han evaluado varios absorbentes. En este trabajo se trata más detalladamente de los problemas de índole especial que aparecen en reactores civiles de tipo Calder y se describen experimentos en los cuales se han hecho ensayos con carbón, cobre, bronce y plata como absorbentes.

INDIAN POINT

Con. Edison's Thorium Converter

APART from detail design, the Indian Point power station of the Consolidated Edison Company of N.Y., Inc., has two main features of outstanding interest. The reactor is a thorium converter, and a considerable proportion of the output of the plant is obtained from fossil-fired superheat.

Indian Point is located on the Hudson River, some 24 miles north of New York City limits. It was the first station for which an AEC construction permit was issued, and construction proper commenced early in 1958. Criticality is expected before the end of the year, and full power operation will commence in the spring of 1962. Construction work is being carried out by Con. Edison, with the Vitro Engineering Company (a division of Vitro Corporation of America) as consultants. The reactor is designed and built by The Babcock & Wilcox Company. The station has a gross electrical output of 275 MW of which 112 MW is produced from oil-fired superheat.

The reactor, which has a gross thermal output of 585 MW, is of the pressurized water type, and the nuclear portion of the plant, including the reactor and its associated steam generators, is enclosed in a steel sphere surrounded by a cylindrical concrete building with a domed roof. On one side of this is the fuel-handling building; on the other, is the "conventional"

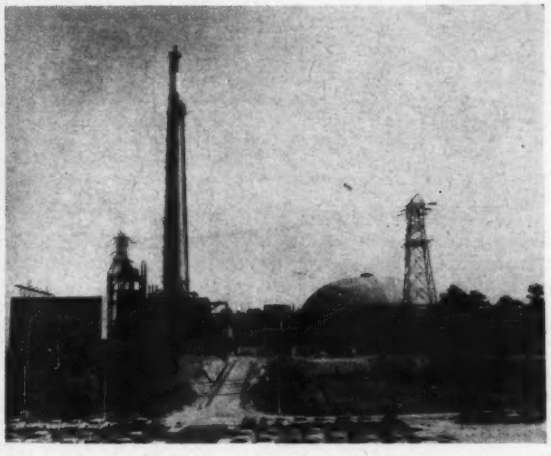


Fig. 2.—General view of station.

plant, including the superheaters and the turbine house. The general arrangement can be seen from the pull-out drawing facing page 420.

Core

In this reactor power is produced by fission of U^{235}, and of U^{233} which is produced during operation from Th^{232} by neutron capture. The fuel is a mixture of highly enriched uranium oxide and thorium oxide and, in order to maintain some uniformity of burn-up, the core is divided into three concentric zones, each having a different uranium-thorium content with the central zone containing the lowest proportion of uranium. The 120 fuel elements are arranged in a square lattice, in rows of 6, 8, 10, 12, 12, 12, 12, 12, 12, 10, 8, 6 elements to form an approximate cylinder, a portion of which

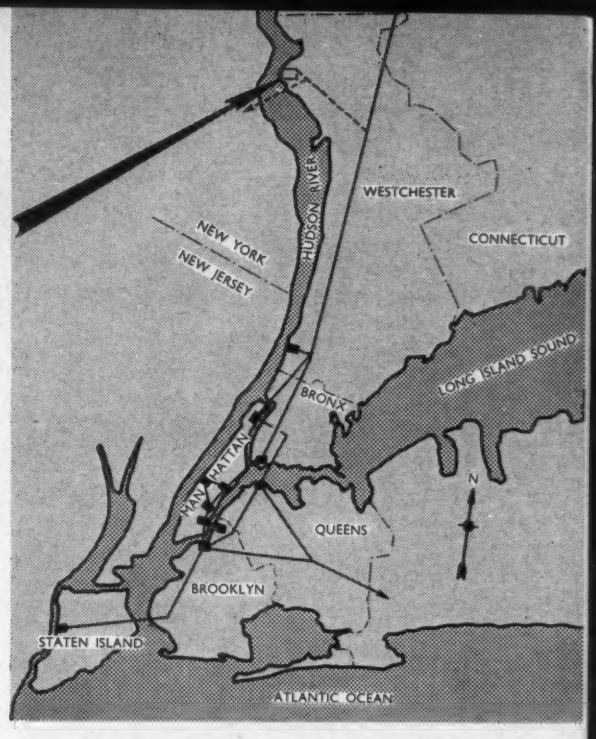


Fig. 1.—Map of New York area, showing location of plant.

is seen in Fig 3. The composition of the fuel at the end of the 600-day core life is given on the pull-out.

A section of the reactor, Fig. 9, shows the core structure to consist of the lower grid plate assembly (comprising the lower grid plate, plenum chamber and control rod guide tubes) which is supported from the reactor vessel; the core shroud, which rests on the lower grid plate and acts as the inner thermal shield; the hold-down columns which surround the fuel elements; the upper grid plate assembly. The last of these, which includes the upper plenum chamber, is held down, against the hydraulic forces of the coolant flow, by the vessel head, and the load is distributed by means of springs, which also allow for differential expansion. Type 304 stainless steel and

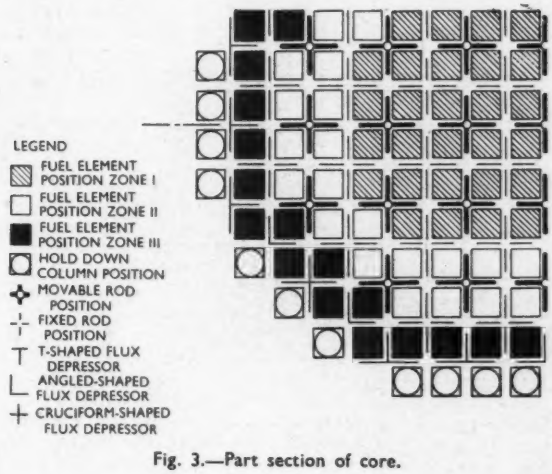


Fig. 3.—Part section of core.

Zircaloy-2 are used for the core components; springs are Inconel-X.

Fuel Elements

The fuel (highly enriched uranium oxide and thorium oxide) is in the form of high-density pellets contained in Type 304 stainless steel tubing to form fuel rods. Washer-type spacers

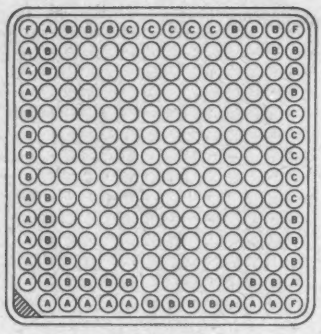


Fig. 4A.—Zone I.
(Blanks are concentration "C".)

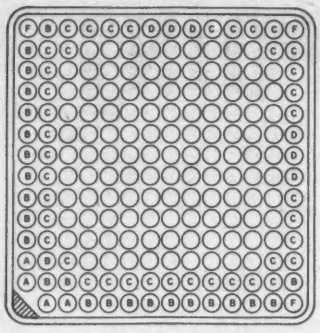


Fig. 4B.—Zone II.
(Blanks are concentration "D".)

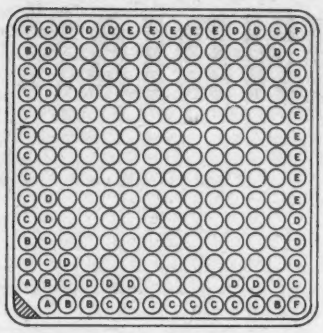


Fig. 4C.—Zone III.
(Blanks are concentration "E".)

Fig. 4.—Key to fuel element loading patterns.
("A" denotes the lowest U^{235} concentration; "E" the highest. "F" rods have 1/3 ThO_2 plus 2/3 concentration "A".)

are located in each rod forming separate compartments, and helium is introduced to fill the spaces.

Stainless steel ferrules, at 9 in intervals, guard against distortion and maintain the rod to rod and rod to outer can spacing. The rods are arranged in a 14 by 14 array, with one corner rod omitted; each bundle, consisting of 195 rods, is brazed up together, and contained in a Zircaloy-2 can of square cross-section, with radiused corners.

As previously mentioned, the core is zoned, to reduce power peaking. The central zone has 32 elements and the two outer zones 44 elements each. The total amounts of U^{235} in the respective zones are 216, 387 and 497 kg. In order to reduce local flux peaking, however, individual rods are graded into six ratios of U-Th content. The arrangement of rods within each element for the three zones is shown in Fig. 4.

This requirement has necessitated rigid control during manufacture, to ensure proper orientation of the bundle pattern within the element. Correct orientation of the element bundle within the can is ensured by the missing corner tube in the bundle, and an index strip in the can.

Control

Control of the reactor is by means of moving control rods; fixed shim rods; fixed flux depressors; burnable poison in the fuel rods; soluble poison in the coolant.

There are 21 hafnium control rods of cruciform section which fit into the spaces between the fuel elements; they are attached to Zircaloy-2 follower rods and are operated from below the reactor. The rods, which are 10 in wide × 0·3 in thick, have a control worth (Δk_{eff}) of 11·2% at 68°F, 13·6% at 450°F and 14% at 503°F. The rod worths were obtained by extrapolation of critical experiments using two-dimensional four group calculations, and are believed to be constant within a margin of −1% and +4% for all possible fixed rod patterns.

Soluble poison in the coolant is used to compensate for the change in reactivity between cold (68°F (20°C)) and the zero power temperature of 450°F (232·2°C), and fixed shim rods will adjust for possible variations from the expected core reactivity.

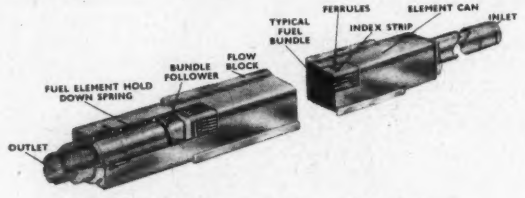


Fig. 5.—General construction of fuel element.

The shim rods, of boron stainless steel, are of cruciform section, 10 in × 0·125 in thick, and the steel contains about 1 w/o boron enriched to about 80% in the B^{10} isotope. The rods are hung from the top transition pieces of the four adjacent fuel elements.

The initial distribution of control between control rods, shims, and soluble poison is given in Table 1.

Table 1. Initial Reactivity Control Distribution (%Δk_{eff})

Effect	Moving rods	Fixed rods	Soluble poison	Total
Shim	4·4	1·0	—	5·4
Temperature (68°–450°F)	—	—	2·8	2·8
Temperature (450°–501°F)	0·6	—	—	0·6
Power Doppler	2·1	—	—	2·1
Eq. Xenon	2·3	—	—	2·3
Eq. Sm	0·7	—	—	0·7
Total	10·1	1·0	2·8	13·9

The maximum worth of the movable rods is required with a clean core at zero power (450°F) when the core will have a maximum reactivity of 14·1%; i.e., 11·1% plus 3% allowance for uncertainties. In this case twelve fixed shim rods will be used, which will total, with the movable rods, a minimum of 16·7%. If the minimum worth of the control rods is taken as 12·6% (i.e., 13·6% −1% allowance for uncertainties) the

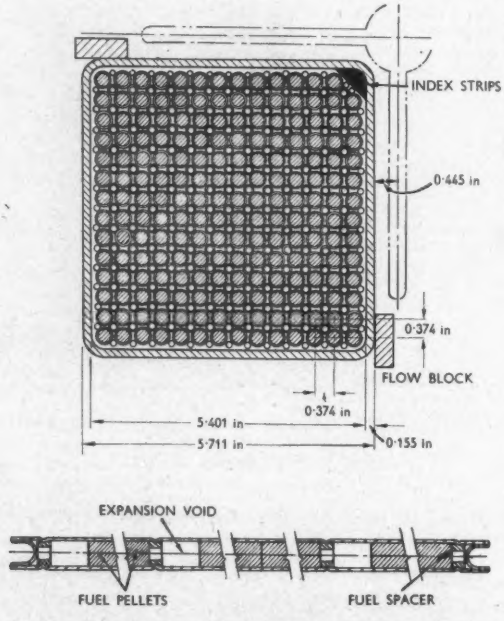


Fig. 6.—Cross section of fuel element and part longitudinal section of fuel rod.

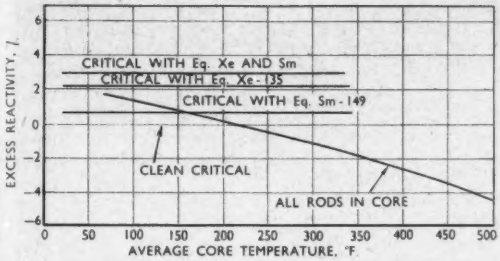


Fig. 7.—Variation of core reactivity with temperature.

12 shim rods will be worth 4·1% with all the moving control rods in the core. The shut-down margin for this worst case is 2·6% k_{eff} and the reactor will be sub-critical with any two control rods fully withdrawn (or an equivalent worth of a larger number of rods).

The excess reactivity of the core as a function of temperature (with all control rods in) is shown in Fig. 7. Without soluble poison, and with equilibrium Sm, the reactor may be cooled to 150°F (65·5°C) without going critical; with equilibrium Xe, criticality will not be achieved at any attainable temperature, with all rods in.

To increase core life by equalizing burn-up, the fuel cladding contains 200-250 ppm of boron, the initial worth of which is some −2·1% to −2·4%.

Water channels not occupied by either control or shim rods at the periphery of the core, are filled with Type 304 stainless steel flux depressor plates, to avoid localized power peaking in the fuel rods adjacent to the channels.

Fig. 8.—View of plant, from reactor end, during construction.

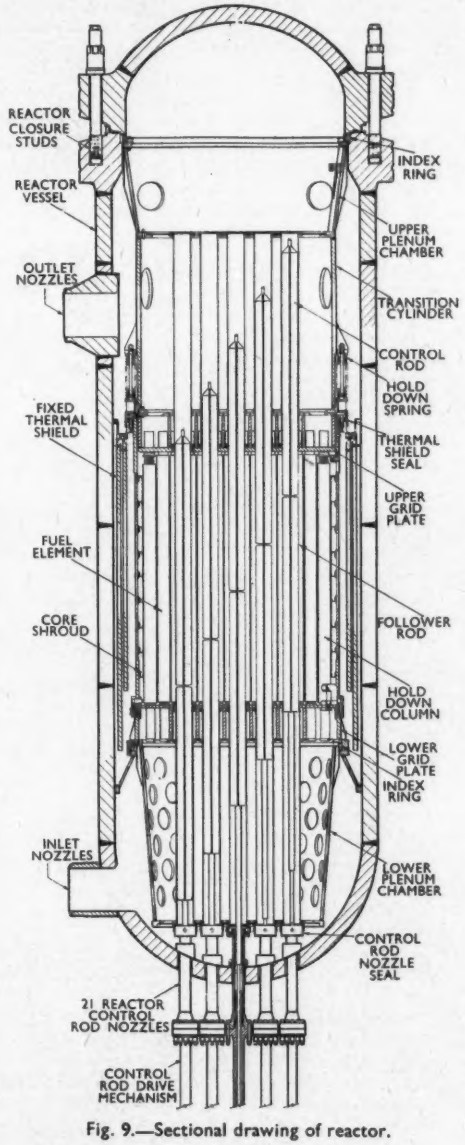


Fig. 9.—Sectional drawing of reactor.

Control Rod Drives

In general principles, the control rod drives resemble those developed for the N.S. *Savannah* (*Nuclear Engineering*, October, 1960, p. 452) in that the rods are moved by hydraulic power, and located by an electrically driven carriage. There is, however, a very large difference in that the rods move upwards from below and, therefore, instead of having to overcome the reactor pressure when inserting the rods for scram purposes, the piston action of the reactor pressure on the rods actually provides the force necessary for scramming, and it is necessary to overcome the reactor pressure for rod withdrawal.

Fig. 11 shows a simplified arrangement of the drive mechanism. Beneath the control rods there is a zirconium follower of the same cross-section, to which is attached the operating shaft which leaves the pressure vessel through a nozzle, and passes through snubber and buffer seal units before being attached to the operating mechanism.

A hydraulic cylinder, with a piston stroke equal to the travel of the control rod, provides the operating force to move the control rod against the operating pressure of the reactor. Its travel is restrained by a coupling spacer which comes into contact with the underside of a carriage moved by two Acme screws which, coupled together by sprockets and chain, are driven through reduction gearing by a motor whose movement is dictated by the control circuits. The inwards movement of the shaft (withdrawing control rods from the core) is effected by the hydraulic pressure on the piston, which is automatically varied to be maintained about 50 psi above the reactor pressure, under the restraint of the nut carriage; outwards movement of the shaft (control rod insertion) is by means of the nut carriage, which forces the shaft downwards against the cylinder pressure. A balancing pressure of determinate value is applied to the top of the piston, to minimize the load on the lead screws. For scramming purposes, the hydraulic pressure on the underside of the piston is released, and the rods are forced outwards by the hydraulic pressure in the reactor.

During scram the speed of the rods is controlled to prevent mechanical damage. For the major portion of the stroke, the speed is kept to less than 16 ft/s by a fixed orifice in the oil

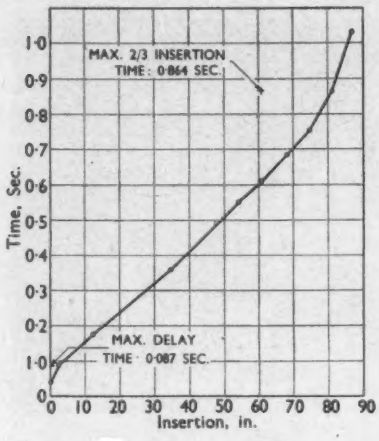


Fig. 10.—Control rod scram performance.

dump line. At about 34 in from the end of the stroke, the snubber unit immediately below the reactor comes into action, when a piston enters a conical bushing, producing very high pressure (2 700 psi) and decelerating the rod to about 0·4 ft/s; when within 7 in of the end of the stroke, the oil snubber mechanism consisting of a dashpot, assisted by a needle valve in the exhaust of the hydraulic cylinder, takes over. A typical scram curve, obtained during prototype tests, is shown in Fig. 10.

Escape of water from the reactor is prevented by the buffer seal unit, which contains 26 seal rings. Clean demineralized water is injected into the seal at the fourth ring from the reactor, at a pressure of about 50 psi above reactor pressure, and about 0·5 gal/min leaks into the reactor; about 1·5 gal/min leaks through the remaining 22 rings and is drained away at atmospheric pressure.

An extensive development programme was carried out on these drives, in three parts. The first part, prototype testing, included the building of a complete rig owned and operated by The Babcock & Wilcox Company, but located at General Electric's San José plant. A test vessel, 20 in i.d. and 30 ft long, surmounted the mechanism to represent a portion of the reactor vessel, and was fitted with dummy reactor internals and provided with a flow of coolant to simulate actual working conditions. The internal guides were arranged so that specified amounts of misalignment could be applied, as required, to enable the effects of core distortion to be studied.

Prototype tests were carried out from June, 1959 to April, 1960, enabling operation in all conditions to be studied, and checks made on materials design and fabrication, and the wearing qualities to be evaluated. The series included a total operation of 56 500 ft, and 500 scrams.

The second and third parts, testing of the production mechanisms, and prototype and production testing on the buffer seals, were carried out by General Electric and the Stein Seal Company of Philadelphia, respectively.

Pressure Vessel

The reactor pressure vessel is 9 ft 9 in i.d., and has an overall height of nearly 40 ft. The shell is constructed of SA212 Grade B fine-grain carbon steel, $6\frac{1}{16}$ in thick, clad with Type 304 stainless steel 0·109 in thick, and the design was in accordance with the ASME Boiler and Pressure Vessel Code (Section 1) for conditions of 1 800 psi and 650°F. Four inlet stubs, $20\frac{1}{8}$ in i.d., are located in the lower part of the shell, and four outlet stubs, of the some diameter, in the upper part. These inlet and outlet stubs are not uniformly distributed around the periphery of the vessel, but are placed in pairs, the angle between the centre lines of each pair being approximately 110°.

The head, which has the same thickness as the shell, is secured by 50 bolts, each $5\frac{1}{2}$ in in diameter, and flange sealing is provided by two metal O-rings. A flexible membrane is welded to both flanges, in addition.

The vessel weighs 255 ton (U.S.) empty, and with water and core internal fittings ($43\frac{1}{2}$ ton) $374\frac{1}{2}$ ton.

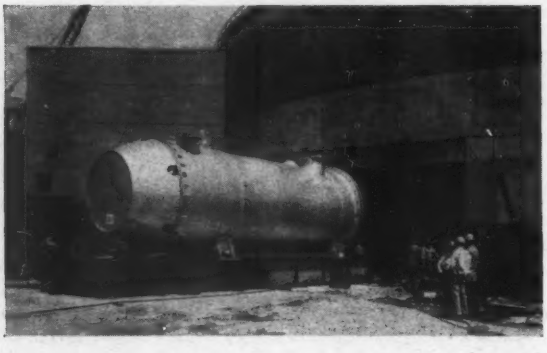


Fig. 12 (left).—Pressure vessel entering reactor building.

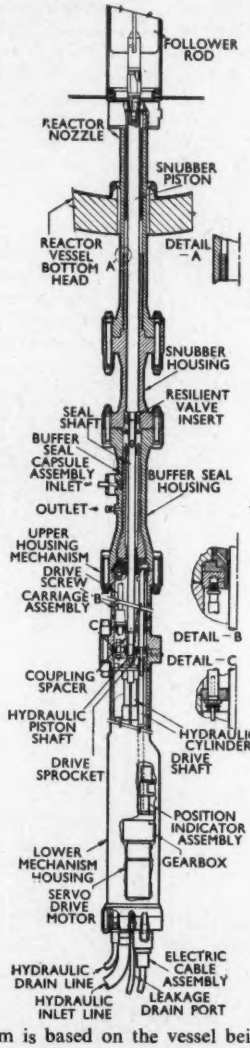


Fig. 11 (right).—Simplified drawing of control rod mechanism.

The design of the system is based on the vessel being fixed, all thermal expansion of the primary system being accommodated by the main coolant piping and the boiler supports.

Primary Coolant System

Water is circulated in four coolant loops, each provided with a separate heat exchanger, steam generator, or boiler, the total flow rate being $52{\cdot}8 \times 10^6$ lb/h at a gauge pressure of 1 485 psi and an average temperature of 502·8°F.

Each loop has two single-speed vertical canned-rotor pumps

operating in parallel, each pump having a capacity of 16 400 gal/min at a differential pressure head of 352 ft. The pumps are mounted on the boilers, and have 18 in outlets. Downstream of each pump is a swing check valve to prevent substantial backflow in case of pump failure, although a definite small flow is allowed even with the valve closed, to prevent thermal shock.

Each loop has two gate valves located as near as possible to the reactor. These are not intended for flow control, their function being to isolate the loop from the main system in case of trouble, and the valve on the reactor outlet closes in 30 s for this purpose. The valve on the reactor inlet is slower in operation, taking 4 min.

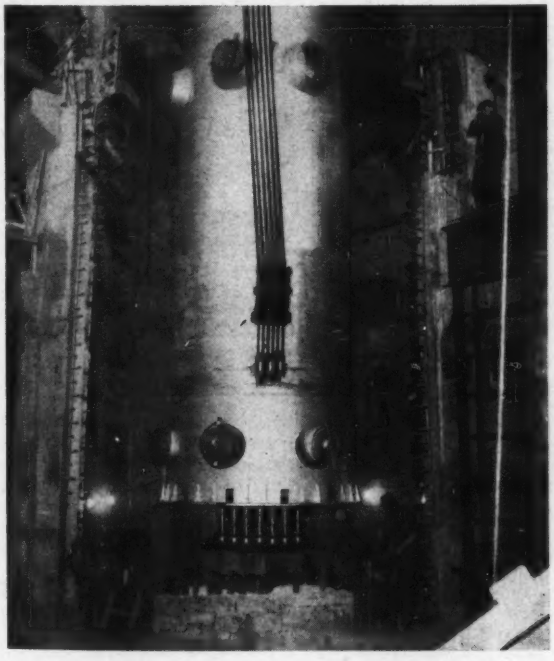
The gate valves are of the split-wedge disc type, in cast stainless steel, and are fitted with buffer seals in the stuffing boxes, with leak-offs to prevent leakage of the coolant to atmosphere, and consequent flashing. Each valve is fitted with a by-pass line and globe valve, for filling and warming-up purposes.

The primary coolant system has a total volume of 4 760 ft^3 (232 948 lb) including the pressurizer. All main piping, in 304 stainless steel, has an o.d. of 24 in, except for the parallel loops through the pumps, which is 18 in. The piping is ultrasonically tested, welds being X-rayed and both piping and welds are dye-penetrant inspected. A check analysis was made on the steel from each heat, from each length of pipe and on each fitting.

Pressurizer

The pressurizer vessel is 6 ft 8½ in o.d., and has a hemispherical top and an ellipsoidal bottom, the overall length being 18 ft, with a skirt extension at the bottom, 3 ft high, and is designed on conventional lines, with water at saturation temperature in the lower portion, and saturated steam above, to act as a cushion.

A 6 in surge line connects the bottom of the pressurizer with the reactor outlet, and permits water to flow either into, or out of, the system following change of volume. The pressurizer is fitted with 490 kW of immersion heaters, arranged horizontally, for pressure raising; pressure reduction is by means of a spray nozzle located in the head of the vessel, fed by a spray line, with a limiting orifice, from the inlet leg of one primary loop between the reactor and the gate valve. The nozzle distributes the water in a conical pattern which does not strike the pressurizer walls. The spray line control valve



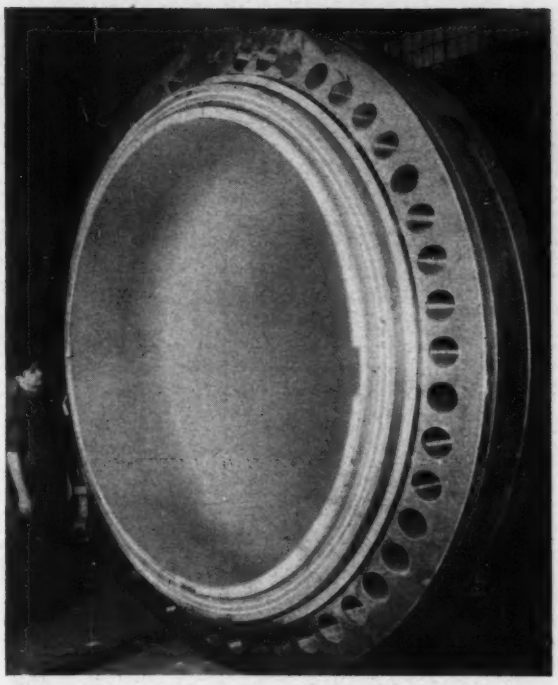


Fig. 13.—Head of pressure vessel, showing grooves for seals.
(These O-rings, by the Advanced Products Co., are believed to be the largest yet manufactured.)

is by-passed by a small orifice line to allow a small constant flow, to prevent either the spray line or the surge line from cooling during prolonged periods of steady load.

Water level in the pressurizer is controlled by regulating the intake of water from the make-up system, and the outflow to the blow-down system. Analysis of system transients shows that the pressurizer system is adequate to accommodate the maximum volume changes resulting from transients up to a step change of 50 MW in reactor output.

Boilers

The steam generator, or boiler, in each loop consists of a U-tube heat exchanger with a single steam and water drum, the overall dimensions of the unit being approximately 41 ft long x 22 ft high x 13 ft wide. The lower drum contains 811

Fig. 14 (above).—Vessel head, with Diamond stud tensioners in position for bolting down.

Fig. 15 (left).—Pressure vessel being lowered into position.

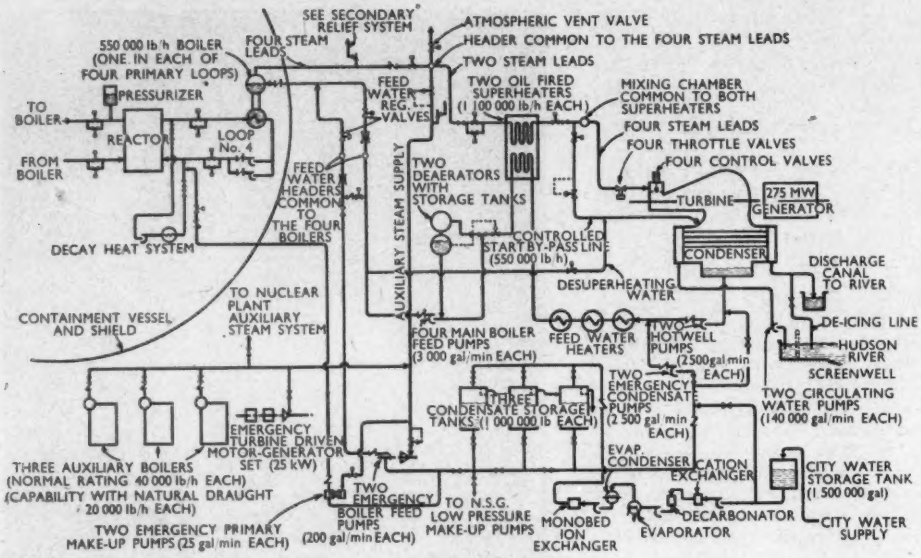

SEE SECONDARY RELIEF SYSTEM
ATMOSPHERIC VENT VALVE
HEADER COMMON TO THE FOUR STEAM LEADS
550 000 lb/h BOILER (ONE IN EACH OF FOUR PRIMARY LOOPS)
FOUR STEAM LEADS
TWO STEAM LEADS
PRESSURIZER
FEED WATER REG. VALVES
TWO OIL FIRED SUPERHEATERS (1 100 000 lb/h EACH)
MIXING CHAMBER COMMON TO BOTH SUPERHEATERS
TO BOILER
REACTOR
LOOP No. 4
FOUR STEAM LEADS
FOUR THROTTLE VALVES
FOUR CONTROL VALVES
FROM BOILER
FEED WATER HEADERS COMMON TO THE FOUR BOILERS
TWO DEAERATORS WITH STORAGE TANKS
TURBINE
275 MW GENERATOR
AUXILIARY STEAM SUPPLY
DECAY HEAT SYSTEM
CONTROLLED START-BY-PASS LINE (550 000 lb/h)
CONDENSER
DISCHARGE CANAL TO RIVER
DESUPERHEATING WATER
TO NUCLEAR PLANT AUXILIARY STEAM SYSTEM
CONTAINMENT VESSEL AND SHIELD
FOUR MAIN BOILER FEED PUMPS (3 000 gal/min EACH)
FEED WATER HEATERS
TWO HOTWELL PUMPS (2500gal/min EACH)
DE-ICING LINE
HUDSON RIVER
SCREENWELL
TWO EMERGENCY CONDENSATE PUMPS (2 500 gal/min EACH)
THREE CONDENSATE STORAGE TANKS (1 000 000 lb EACH)
TWO CIRCULATING WATER PUMPS (140 000 gal/min EACH)
EMERGENCY TURBINE DRIVEN MOTOR-GENERATOR SET (25 kW)
EVAP. CONDENSER
CATION EXCHANGER
CITY WATER STORAGE TANK (1 500 000 gal)
THREE AUXILIARY BOILERS (NORMAL RATING 40 000 lb/h EACH) (CAPABILITY WITH NATURAL DRAUGHT 20 000 lb/h EACH)
TWO EMERGENCY BOILER FEED PUMPS (200 gal/min EACH)
TO N.S.G. LOW PRESSURE MAKE-UP PUMPS
MONOBED ION EXCHANGER
DECARBONATOR EVAPORATOR
CITY WATER SUPPLY
TWO EMERGENCY PRIMARY MAKE-UP PUMPS (25 gal/min EACH)

Fig. 16.—Simplified diagram of plant.

Type 304 stainless steel U-tubes, 1 in o.d. with a wall thickness of 0·062 in, making a total heat transfer area of 13 773 ft², giving a design secondary steam output of 550 000 lb/h at 405 psi, saturated (449°F), from the upper drum, which is connected to the shell of the heat exchanger by risers and downcomers to give adequate natural circulation.

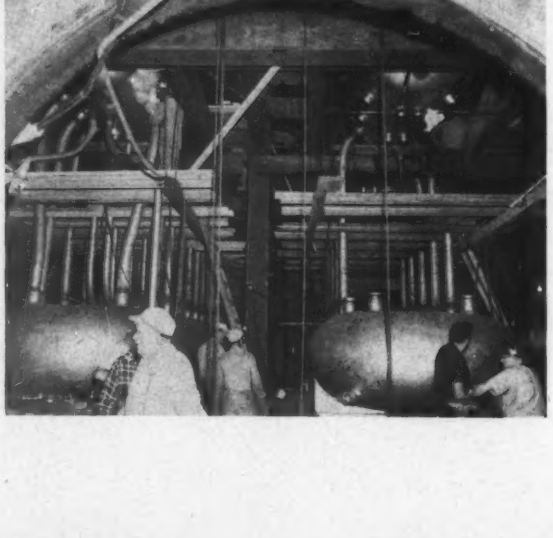
The design of the boilers enables them to withstand load change at 11% per minute indefinitely, and to endure 1 000 cycles of emergency transient conditions of an instantaneous increase of 22% full power at any level between 25% and 95% full power, or a power decrease from 100% to 2% full power in 10 sec.

SECONDARY CIRCUIT

In the secondary circuit are included all the so-called "conventional" items of the plant from the outlet of the main boilers, where two 10 in steam pipes connect to a 20 in pipe carrying the steam from one boiler. The four 20 in pipes that leave the containment sphere are fitted with motor-operated stop valves interlocked with the primary loop isolation valves, so that simultaneous isolation on both primary and secondary sides is possible at will. Piston-controlled solenoid-operated check valves are also included to deal with reverse steam flow, in the case of a major piping leak inside the containment. The four steam mains are cross-connected to two 24 in headers to the superheaters so that any boiler may feed either superheater.

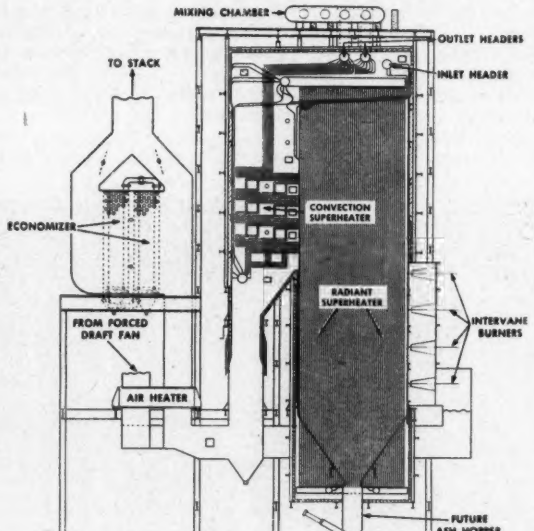
Fig. 17.—(Right) Simplified section of Foster-Wheeler superheater.

Fig. 18.—(Below) Installing the heat exchangers.



Superheaters

The superheaters, which closely resemble "straight-through" conventional boilers, are combined radiation-convection type,



oil-fired, with pressurized furnace. Each superheater has a capacity of 1 100 000 lb/h of 405 psi steam, which is superheated from saturation temperature to 1 000°F.

The radiant portion of the superheater consists of a tangent-tube furnace; the convection section is located in the gas stream just outside the furnace. Ljungstrom air heaters are provided in conjunction with the mechanical-atomization oil burners, together with a forced draught fan; the use of furnace pressurization renders induced-draught fans unnecessary. The design is such that future conversion to pulverized coal firing is possible, if this should be required.

The furnace exit gas temperature is specified as 1 900°F maximum; in addition to the air heater, maximum fuel utilization is obtained by means of an extended-surface economizer, connected in the boiler feedwater supply. With a gas exit temperature of 350°F at the economizer outlet, the guaranteed efficiency of the superheater is 85·5%.

Turbo-Alternator

The turbine is a single-shaft 1 800 rev/min tandem-compound unit rated at 275 MW at rated steam conditions, with a double-flow h.p. cylinder and twin double-flow l.p. cylinders exhausting into a twin-shell condenser. There are one h.p. and three l.p. tappings for bled steam feed heating.

The alternator is a 16 500 V 60 c/s unit, hydrogen cooled, rated at 264 MW with hydrogen at 45 psi, and 286·956 MW at 60 psi.

Condenser

The condenser is of the single-pass type, divided into two shells, with isolating plates which divide each half into two

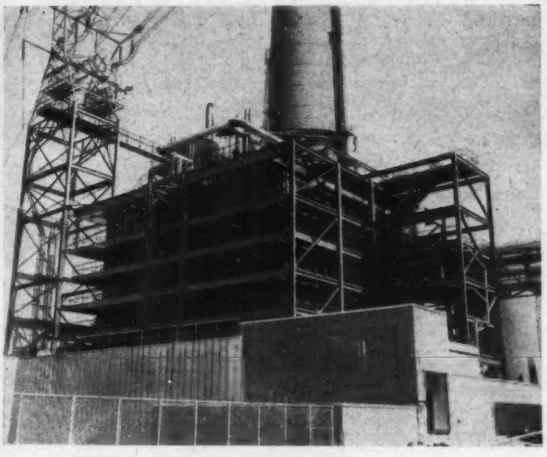


Fig. 20.—General view of superheaters.

zones, each with its own hotwell. Each zone is provided with six condensate collecting boxes arranged in series to cover the whole length of the tubes, and condensate will pass through these boxes before falling into the hotwell, thus permitting careful sampling for detection of tube leaks. Although the specification called for inert gas welding of the aluminium-brass tubes to the silicon-bronze tube plates, even the most minute leaks of cooling water into the condensate cannot be tolerated, particularly at periods of high salt concentration in the Hudson River. Any chloride content in the feedwater would cause stress corrosion in the stainless steel boiler tubes, and monitoring for chloride content is carried out by instruments developed by Con. Edison, based on flame photometry; the instruments are capable of detecting chloride concentrations as low as a few parts per billion (U.S. billion $=10^9$).

In the event of the turbine being shut down and the reactor being left in operation, the condenser is designed to accept steam dumping of about 15% full load, through a turbine by-pass, reducing valve, and de-superheater.

PLANT CONTROL

Automatic control is used above 15% of full load. Control is based upon constant steam pressure, rather than constant reactor temperature, which would have necessitated a reducing valve in the main steam supply line.

The plant controller is designed to permit a maximum pressure variation of ±60 psi, and a maximum power overshoot of 10% under conditions of ramp change of 25 MW/min between 15% and 100% rated power or a 50 MW step change within the plant capacity. It receives, from the non-nuclear instrumentation, a signal representing average steam pressure in

Fig. 19.—Westinghouse turbo-alternator.

all lines and, after comparing this with a reference signal representing the normal pressure, integrates the error to the steam flow signal, to obtain a reactor power demand. Comparison of this with the actual reactor power level signal obtained from the nuclear instrumentation results in a rod velocity demand signal which is proportional to the reactor power change required to meet the requirements of steam flow and pressure.

The 21 reactor control rods are controlled in groups—four of four rods symmetrically arranged around the core, and one group of five, which includes the centre rod. The rod velocity demand signal is transferred from one group to another in programmed sequence; it cannot move more than one group at the same time. Furthermore, only the number of groups required to take the reactor to the rated power will be withdrawn, the remainder of the groups being locked off.

The reactor power level attainable by the plant controller is limited by the flux/flow computer, a duplicated network which co-ordinates the reactor flux level with the flow conditions in the primary loops. With four loops, each containing two pumps, there are 15 flow conditions (including zero flow) which can occur. For each flow condition signals are generated indicating the appropriate maximum power level and the scram power level. The plant controller is unable to demand more power than the limit set; if the power level is already higher than the scram power signal, a reactor scram is initiated through the appropriate scram network, which suspends all other signals and drives all rods inwards at full speed.

FUEL HANDLING

The general arrangement of the fuel handling building is shown on the pull-out drawing, from which it can be seen that an inclined fuel transfer tube penetrates the containment to connect the fuel handling building with the lower section of a pool near the reactor, this being separated from the upper section of the pool by a large valve. The upper section of the pool forms a transfer canal which, when filled with water, covers the entire reactor to a depth of some 20 ft. In normal operation the canal is dry, and the transfer tube, of stainless steel to prevent contamination, is sealed off by a blank flange at its upper end.

When fuel replacement is necessary, the reactor is shut down, and cooled, the sphere is ventilated and monitored, after which the 50-ton gantry crane is inspected and lubricated. With the pools still dry, the access plug is removed and the insulating cover removed from the reactor by the head/grid handling tool, which is supported by a shaft carried by the gantry crane, and is provided with grappling arrangements for attachment to the insulating cover, the reactor vessel head, and the upper core grid. A slot in the floor between the access plug (13 ft 6 in diameter) and the area immediately above the reactor allows passage of the head/grid tool shaft.

After the insulating cover has been removed and placed in its stowage position, a rotary boom hydraulic stud tensioner unit is lowered into position, and all except five or six of the studs are removed. A rotary cutter unit is then used to cut through the seal welds, the remaining studs are removed, and the head is then removed and stowed.

The lower gate valve of the transfer tube (in the fuel handling building) is then closed, the blind flange removed from the upper end of the transfer tube, the head/grid tool is replaced in the access opening and the access plug is replaced.

The sphere pool is then pumped full of borated water, television cameras and lights are installed and the head/grid tool is located over the reactor, and fixed to the upper core grid. This is conveyed under water to a section of the pool known as the reactor internals pit, and is dumped there, with the head/grid tool attached.

The fuel handling tool is then brought into service. This consists of a wheeled carriage carrying a telescopic boom which can grapple fuel elements or control rods, and rotate in either direction to lock or unlock them. By the use of adapters, it can pick up follower rods, snubber pistons, or flux depressor plates and shims. The fuel handling tool travels between a point immediately above the transfer tube to a manipulator, or rotary platform, located immediately above the reactor, and capable of rotating 195° in either direction from the floor slot.

Components such as fuel elements, control rods, etc., are transferred in baskets of stainless steel, of which there are 189; all have the same outside dimensions, but their internal fittings vary to suit the item to be transported. Those for fuel elements, for example, are divided into compartments by cruciform poison strips of boron stainless steel so that they will accommodate four elements. A test programme has shown that the dimensions of the baskets are such that an infinite array could be stored without criticality.

The baskets are located at a basket loading station at an intermediate point between the reactor and the transfer valve. Three operators with remote control consoles and closed-circuit television control transfer operations within the sphere. No. 1 picks up the fuel element and transfers the machine to No. 2, who places it in a basket and returns the machine to No. 1; interlocks prevent both operators having control of the machine at the same time. Having received two elements at diagonal corners, the basket is rotated through 90° and two more are inserted. The machine picks up the basket and transfers it to Operator No. 3, who lowers it into a transfer carriage which travels within the transfer tube and can be up-ended at either end.

The transfer carriage is then lowered into the tube, and the upper gate valve closed. The lower gate valve, in the fuel handling building, is then opened, the carriage is lowered into the fuel transfer pool in the fuel building, up-ended, and the basket transferred, under water, to a storage rack by means of the fuel handling crane; an empty basket is returned to the sphere. Again, all operations are interlocked so that No. 3 operator in the sphere cannot obtain control of the carriage prematurely.

CONTAINMENT AND SHIELDING

The spherical containment vessel is 160 ft diameter, constructed of carbon steel approximately $\frac{3}{4}$ in thick, and designed to withstand internal and external pressures of 25 psi and 1 psi, respectively. The maximum internal pressure is based on the simultaneous release of the contents of the entire primary coolant circuit, and the secondary content of one boiler. The maximum external pressure of 1 psi (differential) will take into account combined changes in barometric pressure and decrease in the internal temperature of the sphere.

The sphere is sunk into the ground for almost half its depth; its equator is, in actual fact, 4 ft 6 in above ground level.

Indian Point is remarkable in that, unlike most reactors previously described, it does not contain the total biological shielding within the sphere. From the centre of the core, the concentric layers of shielding are as follows:—

Steel thermal shield, 1 in thick;
Two similar shields $1\frac{1}{4}$ in thick;
Pressure vessel;
Annular steel tank giving 4 ft 1 in thickness of water;
Concrete structural shell 3 ft thick;
Containment sphere;
Annular biological shield of concrete 5 ft 6 in thick.

The shielding inside the sphere is sufficient to ensure that the sphere can be occupied, and maintenance carried out only when the reactor is shut down.

The annular water shield is designed to reduce the neutron flux from 10^9 to 10^4 n/cm^{2}s; it also attenuates the core gamma during operation and the fission product gamma during shut-down.

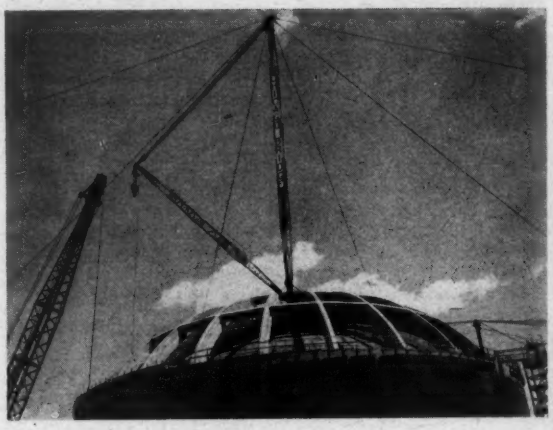


Fig. 21.—Outer biological shield, during roof construction.

The 3 ft inner shield immediately surrounding the water tank will reduce the gamma dose rate during operation to about 700 mrem/h, and attenuate the neutron flux to insignificant levels.

The biological shield within the sphere also includes a 2 ft concrete floor above the reactor at an elevation of 108 ft.

The outer biological shield, surrounding the sphere, is designed to attenuate the core gamma radiation, as well as that from the $O^{16}(n\,p)N^{16}$ water activation from the primary loops, so that the total dose rate is less than 0·75 mrem/h at the outer shield, and 0·057 mrem/h at the boundary of the property (1 500 ft), as well as protecting personnel in the fuel handling and reactor service buildings in the event of the maximum credible accident.

The outer biological shield is 170 ft i.d. It has a self-supporting arched roof some 3 ft thick.

AUXILIARY AND SUPPORTING SYSTEMS

The Indian Point plant requires, for efficient functioning, a large amount of auxiliary equipment for maintaining both primary and secondary water at a high level of purity, for disposal of fission products, and to guard against all foreseeable failures. While much of this is incorporated in the main gear, it is convenient, for clarity, to consider it as a number of separate systems.

Decay Heat Cooling System

A separate cooling system is provided for dealing with the decay heat after reactor shut-down. It is not put into operation until the primary coolant system has been reduced to below 260°F and 200 psi, by exhausting boiler steam to atmosphere or to the condenser, and by operating the water spray in the pressurizer and/or venting the pressurizer steam space.

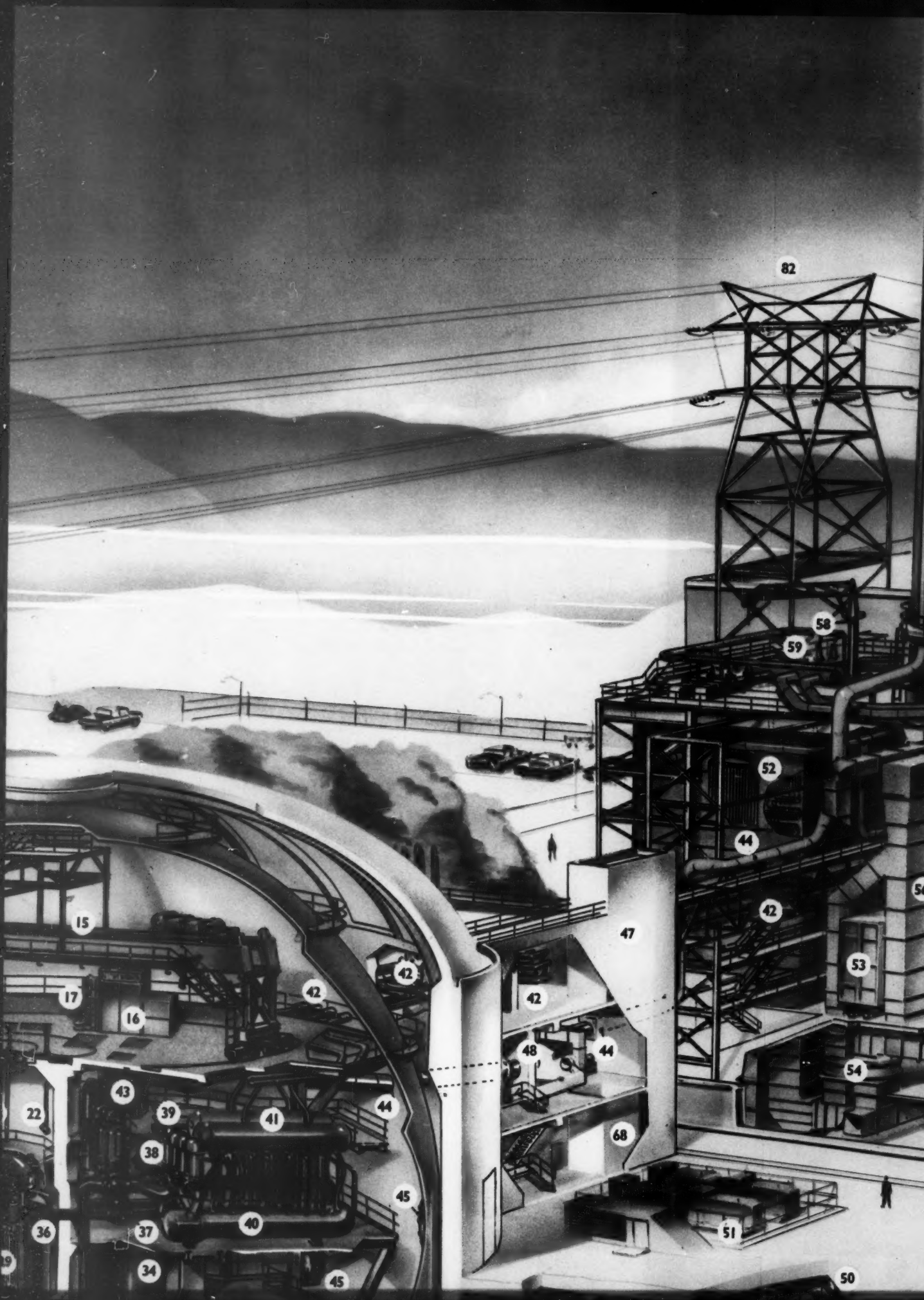
The decay heat system consists of two shell and tube heat exchangers, each having 1 288 ft^2 of heat exchange surface, and designed to remove 20×10^6 Btu/h when primary coolant at 180°F is circulated at 2 400 gal/min. Based on the decay heat from a core irradiated for 600 days at full power, the decay heat system will cool the primary coolant to 140°F in about 27 hours.

Two pumps circulate the primary coolant through each heat exchanger, each pump of a pair being fed from different electrical supplies. The two parallel loops are both connected to the reactor sides of the stop valves on No. 4 primary coolant loop; a by-pass is provided so that the system may be warmed up before being placed in circuit, to avoid reactivity surges due to slugs of cold water.

Blow-down

In order to keep the primary water within acceptable limits of both dissolved solids and radioactivity, a proportion of the

Indian Point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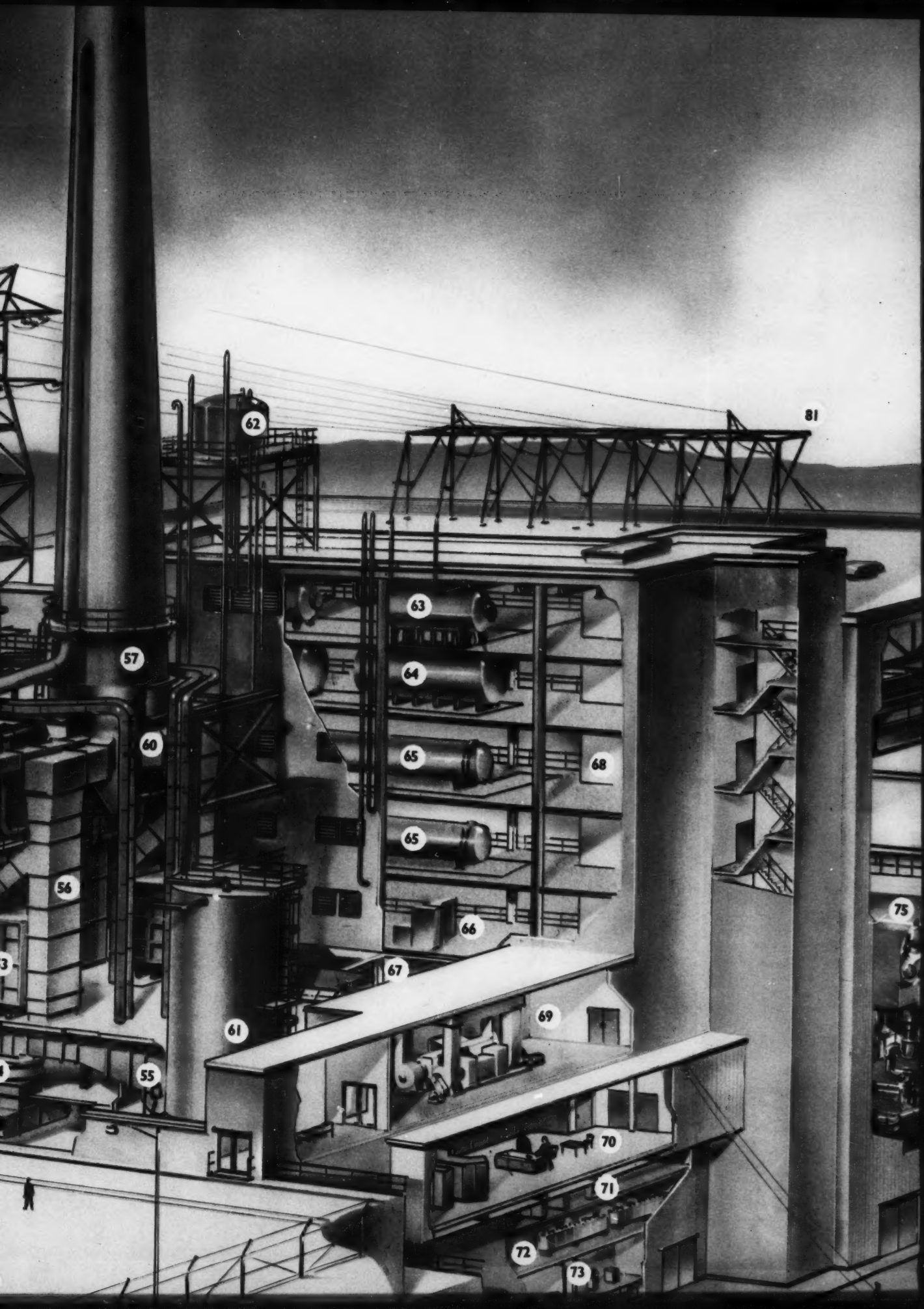
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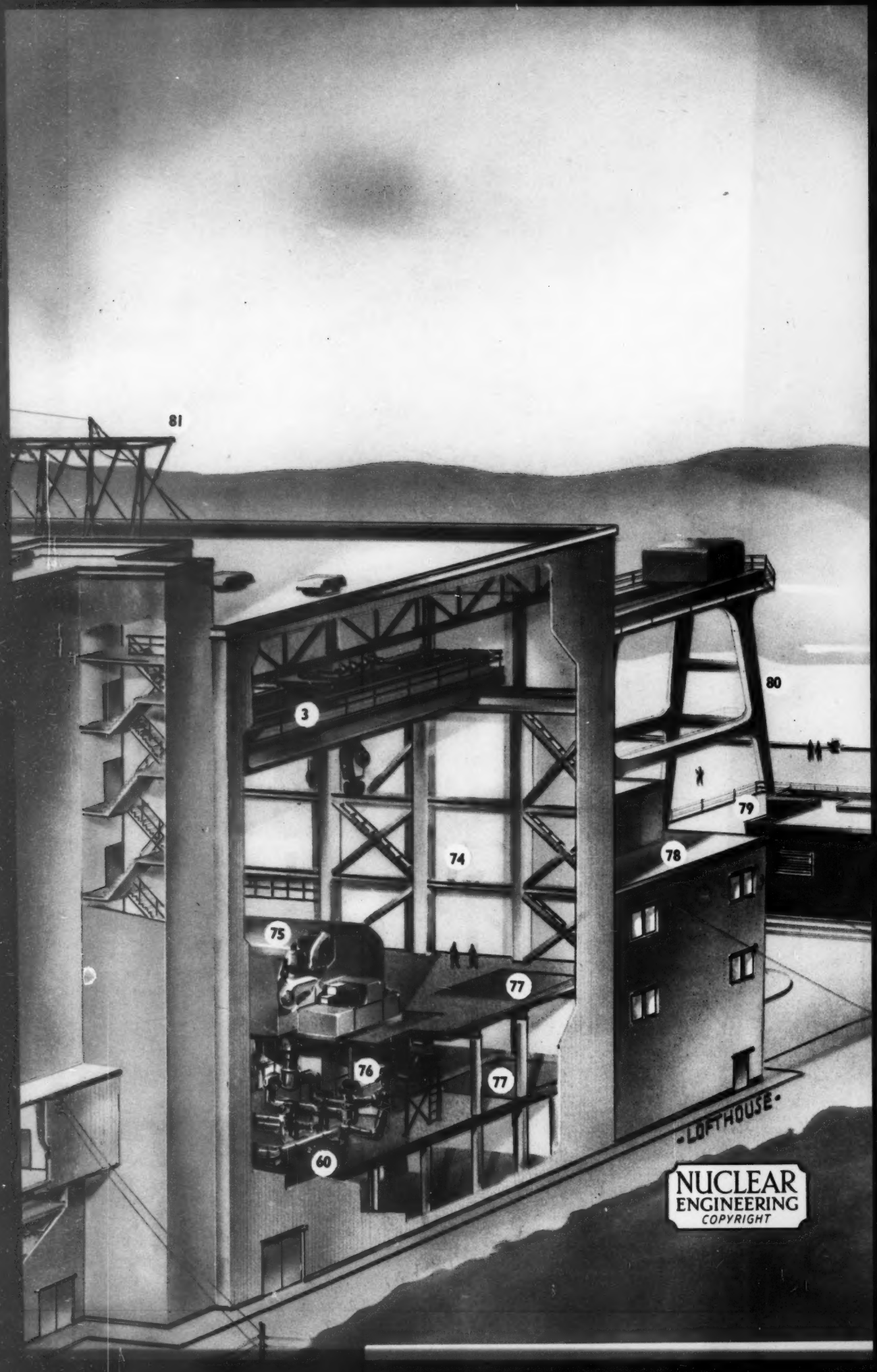
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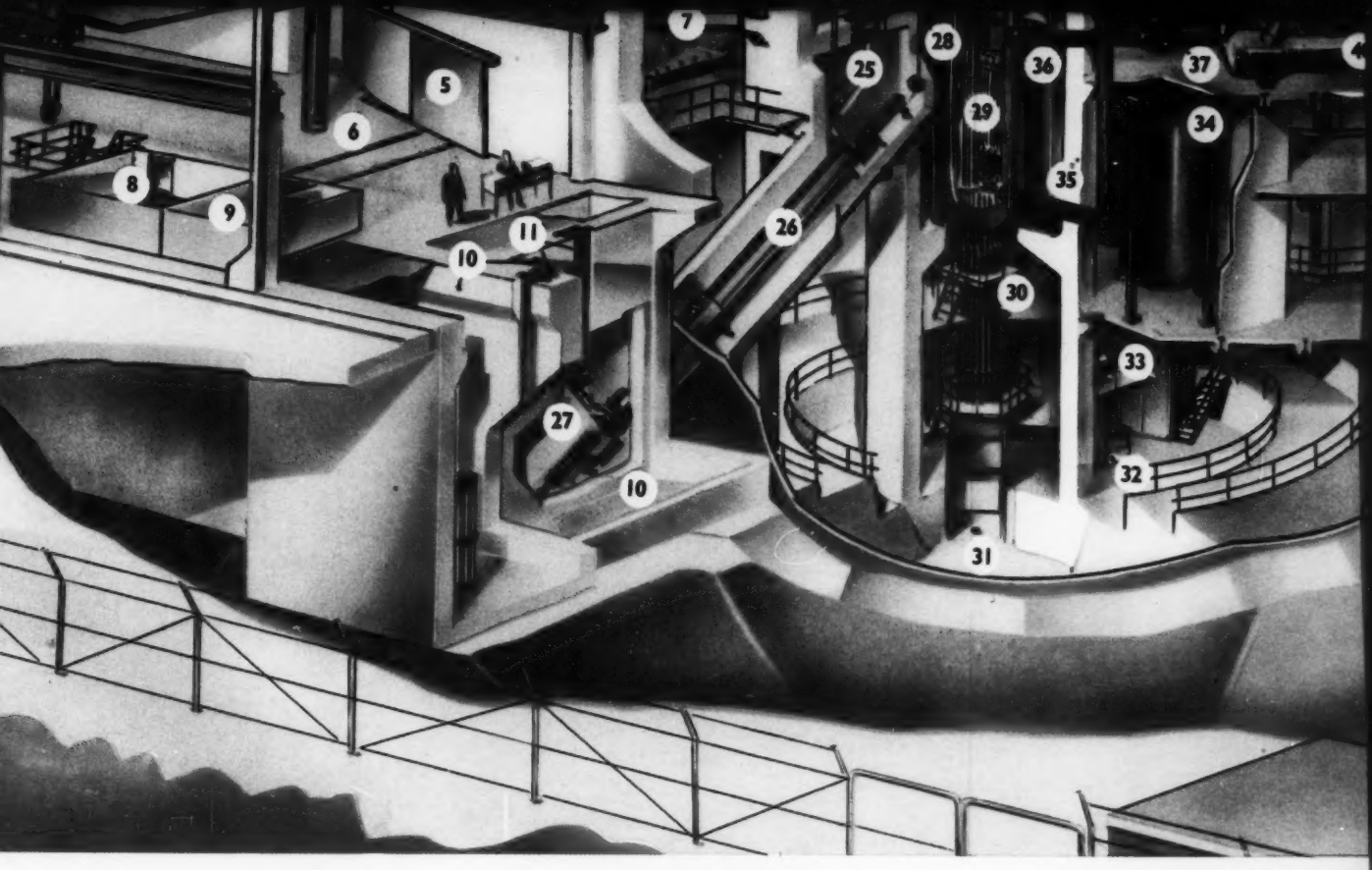
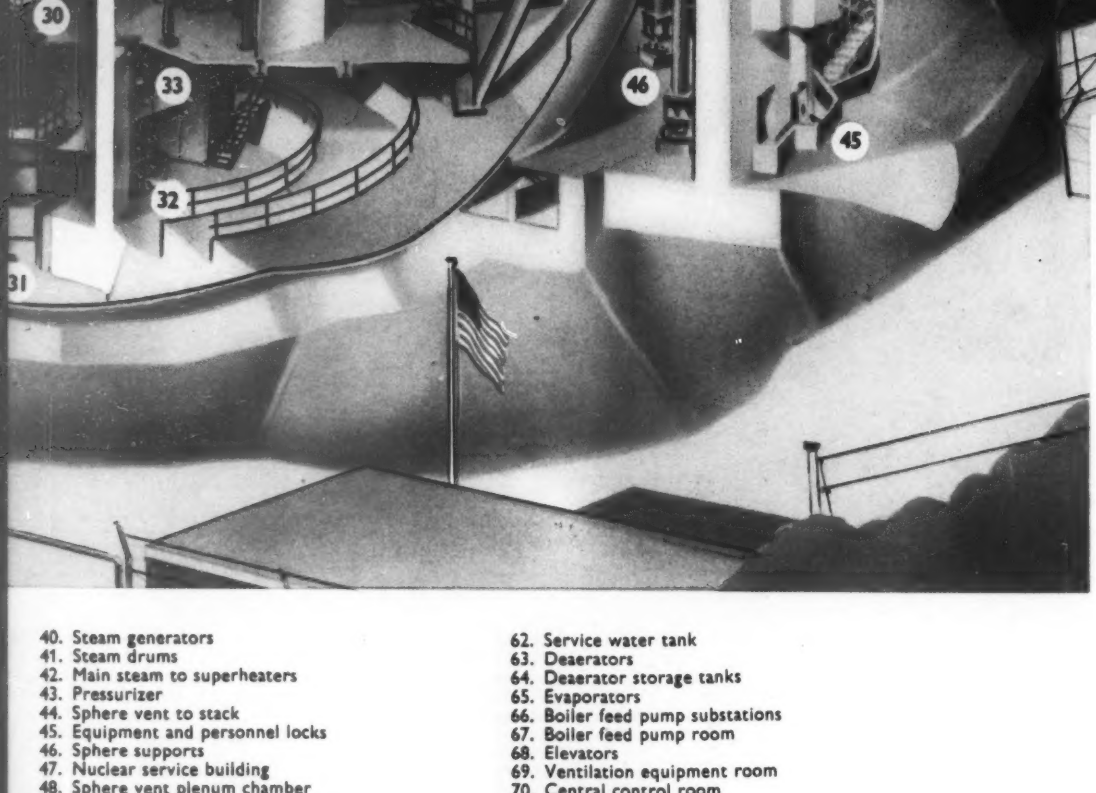


KEY

INDIAN POINT

1. Chemical systems building.
2. Fuel handling building
3. Overhead travelling cranes
4. Equipment hatches
5. Equipment removal shield door
6. Equipment removal rails
7. Equipment removal trolley
8. Decontamination wash pit
9. Decontamination room
10. Fuel transfer pools
11. Water gates
12. Sphere ventilation supply room
13. External radiation shield.
14. Containment sphere
15. Polar crane
16. Elevator
17. Reactor tool shaft storage
18. Refuelling console (normally unattended)
19. Rotating shield plugs
20. Refuelling shuttle hoist
21. Storage pit hatch cover
22. TV cameras (3)
23. Fuel transfer manipulator
24. Transfer tube upper valve
25. Storage pit
26. Transfer tube
27. Transfer tube lower valve and shuttle
28. Reactor vessel
29. Reactor core
30. Control rod drives
31. Reactor pit
32. Sump
33. Sump pump
34. Blowdown tanks (2)
35. Reactor inlet
36. Reactor outlet
37. Gate valves
38. Check valves
39. Pumps
40. Steam generators
41. Steam drums
42. Main steam to superheaters
43. Pressurizer
44. Sphere vent to stack
45. Equipment and personnel locks
46. Sphere supports
47. Nuclear service building
48. Sphere vent plenum chamber
49. Service building ventilation room
50. H_2SO_4 storage tanks
51. Transformer yard
52. Oil fired superheaters (2)
53. Economizers (2)
54. Airheaters (2)
55. Blowers (2)
56. Flues to stack (2)
57. Stack
58. 10-ton jib crane
59. Controlled start piping
60. Superheated steam to turbines
61. Condensate storage tanks (3)



40. Steam generators
41. Steam drums
42. Main steam to superheaters
43. Pressurizer
44. Sphere vent to stack
45. Equipment and personnel locks
46. Sphere supports
47. Nuclear service building
48. Sphere vent plenum chamber
49. Service building ventilation room
50. H_2SO_4 storage tanks
51. Transformer yard
52. Oil fired superheaters (2)
53. Economizers (2)
54. Airheaters (2)
55. Blowers (2)
56. Flues to stack (2)
57. Stack
58. 10-ton jib crane
59. Controlled start piping
60. Superheated steam to turbines
61. Condensate storage tanks (3)
62. Service water tank
63. Deaerators
64. Deaerator storage tanks
65. Evaporators
66. Boiler feed pump substations
67. Boiler feed pump room
68. Elevators
69. Ventilation equipment room
70. Central control room
71. Terminal board room
72. Battery room
73. Water treatment rooms
74. Turbine hall
75. Turbine and generator
76. Main steam stop valves
77. Loading well covers
78. Administration block
79. Screenwell house
80. Gantry crane
81. Take-off tower
82. Transmission towers

OWNER/ BUILDER:	Consoli
NUCLEAR DESIGN:	The Bal
PURPOSE:	Power
LOCATION:	Indian F York C
TYPE:	Therma fired su
PROGRAMME:	Constru Critical Full pow
CAPACITY:	Therma Gross e (163 MV
FUEL:	Enriche Initial lo Fuel pel Claddin Rod, o.d Fuel ele pitch fo rod om Bundle Element Active Max. cl Max. fu Heat tr Mean h Max. he
CORE:	Approx (2·515 m No. of f Square concent Zone lo k_∞: 1·2 k_{eff}: 1·1

R:

Consolidated Edison Company of New York, Inc.

N:

The Babcock & Wilcox Company

Power production.

Indian Point, on the Hudson River, about 24 miles north of New York City.

Thermal, heterogeneous, PWR, thorium converter with fossil-fired superheat.

Construction commenced December, 1957.
Criticality scheduled for late 1961.
Full power operation scheduled for spring of 1962.

Thermal output of reactor: 585 MW
Gross electrical output of station: 275 MW,
(163 MW from reactor, 112 MW from fossil superheat).

CORE:

Enriched uranium oxide (U^{235}) and U^{233} converted from ThO_2.
Initial loading: 1 100 kg U^{235} (1 185 kg U), 17 200 kg ThO_2.
Fuel pellets: 0·26 in (6·6 mm) dia. × 0·8 in (20·3 mm) long.
Cladding: boron-modified type 304 stainless steel tube.
Rod, o.d.: 0·304 in (7·72 mm).
Fuel element comprises: 195 rods at 0·374 in (9·3 mm) square pitch forming a square bundle of 14×14 rods, with one corner rod omitted.
Bundle encased in 0·155 in (3·94 mm) Zircaloy-2.
Element dimensions: 5·711 in (145 mm) square.
Active length: 98·5 in (2·5 m).
Max. cladding surface temp: 319°C.
Max. fuel temp.: 1 927°C.
Heat transfer area: 15 600 ft².
Mean heat flux: 128 000 Btu/ft²h.
Max. heat flux: 533 000 Btu/ft²h.

Approximate cylinder: 6 ft 6 in (1·981 m) equiv. dia. × 8 ft 3 in (2·515 m) high.
No. of fuel elements: 120.
Square lattice: pitch 6·32 in (160·5 mm) arranged in three concentric zones of 32, 44, 44 elements.
Zone loading: 6·75, 8·79 and 11·3 kg U^{235}/element respectively.
k_∞: 1·2.
k_{eff}: 1·14.

Excess reactivity requirements:

Equilibrium Xe	2·3%
Equilibrium Sm	0·7%
Temperature (68–503°F)	3·4%
Power Doppler	2·1%
Fuel burn-up and fission product build-up	5·4%
Total	13·9% $\Delta k/k$

Burn-up: average 18 000 MWd/t; max. 50 000 MWd/t.
Estimated core life: 600 days.
Loading at end of 600 days:

U^{233}	147 kg.
U^{234}	8·6 kg.
U^{235}	695 kg.
U^{236}	76 kg.
Pa^{233}	14 kg.
ThO_2	16 950 kg

Total core volume: 7 740 l.
Average power density: 76 kW/l.
Peak/average power density ratio: 4·2/1
Average thermal neutron flux: $1·5 \times 10^{13}$ n/cm²s.
Conversion ratio: 0·46.

CONTROL:

Cruciform hafnium rods, fixed boron steel shim rods, burnable poison in fuel element cans, and soluble poison in coolant (under cold shut-down conditions).
No. of hafnium rods: 21.
Dimensions: 10 in square × 0·3 in thick.
Total worth (Δk_{eff}) at various temperatures:

68°F	11·2%
450°F	13·6%
503°F	14%

Initial worth of boron (200–225 ppm) in cladding: −2·1% to −2·4%.
Soluble poison worth at 68°F: 2·8%.
Control rods operated by electro-hydraulic mechanisms from beneath the reactor.

COOLANT:

Pressurized light water.
Operating pressure: 1 485 psig.
Inlet temperature: 486·5°F (252°C).
Outlet temperature: 519°F (270·8°C).
Total flow through reactor: $52·8 \times 10^6$ lb/h.
Coolant velocity inside elements: 21·5 ft/s.
Number of coolant loops: 4.
No. of pumps: 8.
Type: vertical centrifugal, canned rotor.

COOLANT

PRESSURE VESSEL:

SHIELDING

CONTAIN

STEAM PL

The World's Reactors

No. 34 Indian Point

Excess reactivity requirements:

Equilibrium Xe	2·3%
Equilibrium Sm	0·7%
Temperature (68–503°F)	3·4%
Power Doppler	2·1%
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Total flow through reactor: $52{\cdot}8 \times 10^6$ lb/h.
Coolant velocity inside elements: 21·5 ft/s.
Number of coolant loops: 4.
No. of pumps: 8.
Type: vertical centrifugal, canned rotor.

COOLANT:

Operating head: 125 psi; capacity: 16 400 gal/min.
Speed: 1 800 rev/min; power: 1 600 h.p.
Coolant piping size (main loop): 24 in.
Total coolant circuit volume (inc. pressurizer): 4 760 ft³.

PRESSURE VESSEL:

Overall height: 39 ft 11½ in, 9 ft 9 in i.d.
Shell thickness: 6-15/16 in.
Material: SA212 Grade B CS fine grain steel.
Clad with 0·109 in type 304 stainless steel.
Design pressure: 1 800 psig.
Design temperature: 650°F.
Head closure: 5½ in bolts (50) with O-ring gasket, and seal weld.
Weight of vessel without core or coolant: 255 ton.

SHIELDING:

Internal thermal shields concentric with core:
two permanent, each 1¾ in thick, one removable, 1 in thick; all in type 304 stainless steel.
Annular shield of light water surrounding vessel, 49 in thick.
Inner biological shield of concrete 3 ft thick, surrounding water shield.
Outer biological shield of concrete 5½ ft thick surrounding containment sphere.

CONTAINMENT:

Sphere: 160 ft (48·768 m) diameter.
Material: carbon steel ⅞ in (22·2 mm) thick.
Design max. internal pressure: 25 psig.
Design max. external/internal differential pressure: 1 psi.

STEAM PLANT:

No. of boilers: 4, shell and tube type, with separate steam drum.
No. of tubes: 811 1 in o.d. × 0·062 in thick, type 304 s.s.
Heat transfer area: 13 773 ft² per boiler.
Design pressure: 1 800 psi (primary).
Design temperature: 650°F (343·3°C).
Design steam flow: 550 000 lb/h.
Steam conditions: 405 psi, saturated (449°F) (231·5°C).
Feed temperature: 330°F (165·6°C).
Superheaters: two, oil-fired type.
Steam conditions at t.s.v.: 370 psia, 1 000°F (537·8°C).
Turbine: 1 800 rev/min tandem compound, double-flow h.p. cylinder, twin double-flow l.p. cylinders, rated 275 MW at rated steam conditions and 1 inHg condenser pressure.
Condenser: 212 000 ft², capable of acting as dump condenser for up to 15% full load, by means of turbine by-pass, reducing valve, and desuperheater.

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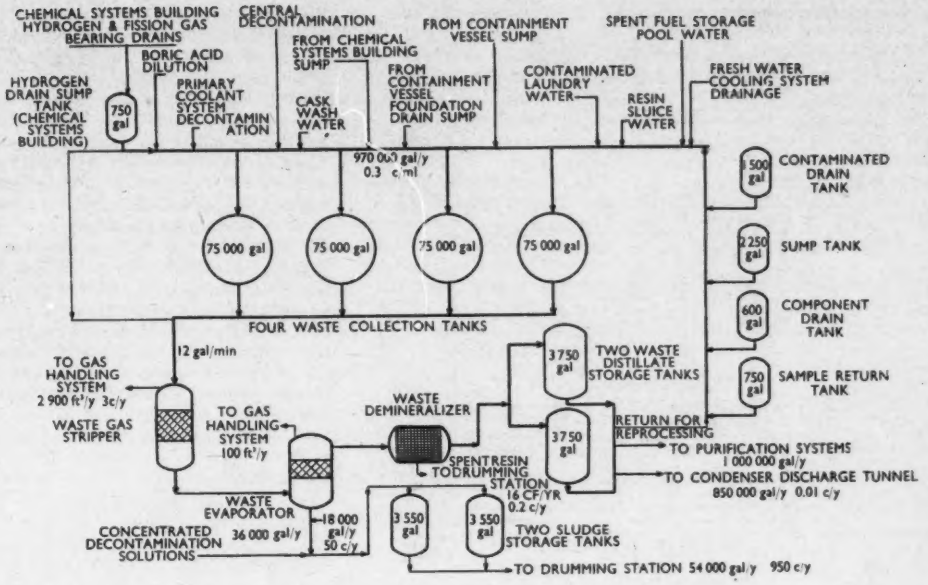
CHEMICAL SYSTEMS BUILDING HYDROGEN & FISSION GAS BEARING DRAINS
CENTRAL DECONTAMINATION
FROM CHEMICAL SYSTEMS BUILDING SUMP
FROM CONTAINMENT VESSEL SUMP
SPENT FUEL STORAGE POOL WATER
BORIC ACID DILUTION
HYDROGEN DRAIN SUMP TANK (CHEMICAL SYSTEMS BUILDING)
750 gal
PRIMARY COOLANT SYSTEM DECONTAMIN ATION
CASK WASH WATER
FROM CONTAINMENT VESSEL FOUNDATION DRAIN SUMP
CONTAMINATED LAUNDRY WATER
RESIN SLUICE WATER
FRESH WATER COOLING SYSTEM DRAINAGE
970 000 gal/y 0.3 c/ml
1 500 gal
CONTAMINATED DRAIN TANK
75 000 gal
75 000 gal
75 000 gal
75 000 gal
2 250 gal
SUMP TANK
600 gal
COMPONENT DRAIN TANK
FOUR WASTE COLLECTION TANKS
12 gal/min
3 750 gal
TWO WASTE DISTILLATE STORAGE TANKS
750 gal
SAMPLE RETURN TANK
TO GAS HANDLING SYSTEM 2 900 ft³/y 3c/y
WASTE GAS STRIPPER
TO GAS HANDLING SYSTEM 100 ft³/y
WASTE DEMINERALIZER
3 750 gal
RETURN FOR REPROCESSING
TO PURIFICATION SYSTEMS 1 000 000 gal/y
SPENT RESIN TO DRUMMING STATION 16 CF/YR 0.2 c/y
TO CONDENSER DISCHARGE TUNNEL 850 000 gal/y 0.01 c/y
WASTE EVAPORATOR
18 000 gal/y 50 c/y
3 550 gal
3 550 gal
TWO SLUDGE STORAGE TANKS
CONCENTRATED DECONTAMINATION SOLUTIONS 36 000 gal/y
TO DRUMMING STATION 54 000 gal/y 950 c/y

Fig. 22.—Simplified diagram of liquid waste disposal system.

primary water is continuously removed and replaced by purified make-up. The blow-down is first cooled from its temperature of 485°F to 200°F (in a regenerative cooler which heats the make-up water) de-pressurized, and stored in the blow-down tanks, where it is temporarily held for decay of short-lived isotopes before being piped to the purification plant.

The equipment is duplicated, and either section can handle up to 80 000 lb/h. It is found, however, that 36 000 lb/h will keep the total solids content of the primary loop below 1 ppm, and the radioactivity equilibrium level at 4·16 μc/ml.

Should neither set of equipment be operative, for any reason, blow-down can be routed through emergency coolers (cooled by a fresh-water circuit which is itself cooled by river water) directly to the purification plant.

Seal Water and Primary Make-up

High pressure clean water is supplied for make-up of primary coolant, for the buffer seals of the control rod drives, and for gland sealing of the stop valves. It is provided by three 200 gal/min multi-stage centrifugal pumps, at 1 700 psi. As previously indicated, a proportion of the water supplied to the seals will enter the reactor, while some will emerge at the end of the seal; this is collected and returned to the clean storage tanks—should monitoring reveal a high radiation level, it is diverted to the blow-down tanks for subsequent purification.

Normal make-up, after cooling the primary blow-down (36 000 lb/h) would be about 26 000 lb/h, assuming that 10 000 lb/h enters the primary circuit through the seals. In cooling the blow-down to 200°F, it is itself heated to 375°F before entering the primary circuit. As more water is required to cool the blow-down than is actually required for make-up, the excess make-up is itself cooled and returned to the clean water storage tanks. If the make-up requirements should exceed the blow-down cooling requirements, additional make-up is provided from a by-pass round the cooler.

In the event of a total failure of all pumps, emergency make-up pumps, driven by steam (either from the main supply or from auxiliary boilers) are available; there are two, each with a capacity of 25 gal/min. These discharge into Loop 4, through the decay heat system piping.

Primary Relief System

Normally, pressure rises in the primary system are dealt with by the pressurizer control system. To deal with abnormal coolant expansion, there is, first of all, a pilot-operated solenoid safety valve designed to operate at 1 675 psi, and to relieve 25 000 lb/h of steam or 280 gal/min of water. Four spring-loaded pop safety valves are also provided, two set at 1 720 psi and two at 1 772 psi, either pair being capable of isolation. In addition, each coolant loop is provided with a spring-loaded relief valve to provide protection when the loop is isolated from the reactor vessel.

All valves discharge through piping to the primary relief tank, which is provided with internal piping and submerged nozzles so that steam is condensed. The tank is of sufficient size to accommodate the combined discharge of all valves for 6½ minutes before the pressure reaches 45 psi; at this point a safety valve opens and discharges to the containment sphere; this is backed up by a rupture disc, bursting at 50 psi.

Primary Vent System

This allows air and other gases to be vented from various portions of the primary system and, when necessary, to supply air during draining of any part of the primary system. Vent lines from individual items are connected to gas collection headers which, in turn, direct the vented gases to the chemical processing system and, if they can be safely discharged without treatment, directly to the stack.

Boron Addition System

As briefly mentioned, negative reactivity is provided at low temperatures, by the use of boron in the coolant. The boron addition system provides a convenient method of adding boron, in the form of boric acid, as required to the coolant and to the water in the fuel transfer and storage canals.

Two steam-heated stainless steel storage tanks contain a total of 500 ft³ of 19% boric acid solution (5 400 lb boric acid) at a temperature of 190°F to prevent re-crystallization; this quantity is sufficient to produce 30 000 ft³ at a concentration of 4 000 ppm. The tanks are also utilized for preparing the solution in the first place, and an electric mixer is permanently installed in each tank.

The boric acid can be added in two ways; by high-pressure pumps at 1 800 psi through the seal water and primary make-up line and through the emergency make-up piping, or, by means of a low-pressure transfer pump, into one of the clean water storage tanks, until this has been borated up to the required concentration. It is estimated that the time taken to borate the entire primary coolant system up to the 4 000 ppm required will be about 8 hours. Removal of the boron will take about the same time and will be accomplished by flushing with clean water to the liquid waste disposal system, requiring about 19 000 ft³ of water. Final clean-up will be accomplished in the

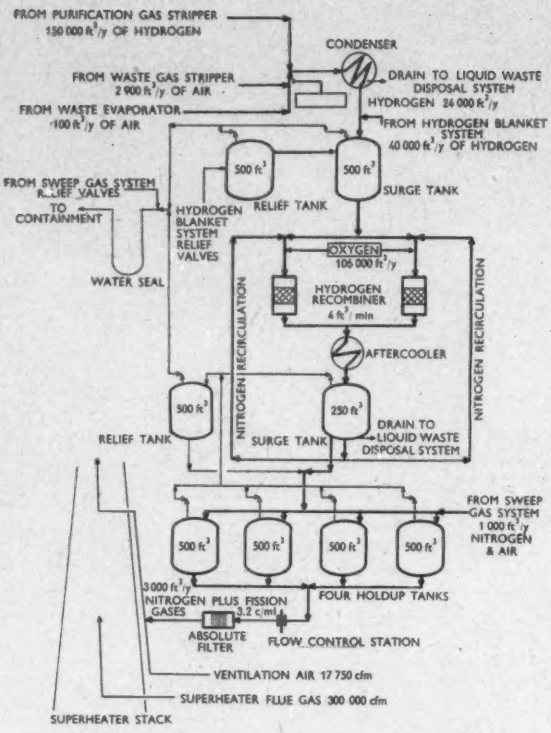

Fig. 23.—Simplified diagram of gaseous waste disposal system.

purification system by two ion-exchangers; about 22 ft³ of resin will be required.

Purification System

Three transfer pumps, each having a capacity of 110 gal/min, and each capable of maintaining the normal blow-down of 36 000 lb/h, are provided for pumping water from the blow-down storage tanks to the purification system, where the first operation is filtration, a bank of eight cellulose filters, of which two are normally in operation, filtering particles down to 5 μm. These filters can be removed and replaced while the system is in operation.

From the filters the water is normally passed to coolers where it is cooled to about 120°F to prevent damage to ion-exchange resins. If necessary, however, it may be diverted to a scrubber unit, for stripping of fission product gas.

The cooled water then passes through a bank of cation exchangers, and a bank of mixed-bed exchangers. When boric acid clean-up is required, the mixed-bed exchangers are by-passed, and the water is routed through anion exchangers. Boric acid solutions stronger than 500 ppm are not tested.

Finally, the water is passed through banks of sintered stainless steel filters (10 μm) to remove any resin fines carried over from the ion exchangers, and stored in three 5 000 ft³ stainless steel tanks, blanketed with hydrogen. The water is introduced through spray nozzles when it is desired to saturate it with hydrogen; through submerged nozzles when it is not.

Auxiliary Steam System

Steam from the nuclear boilers is available at 250 psi through reducing valves, and this supply is interconnected with three 40 000 lb/h auxiliary boilers. Steam drive is provided for the emergency boiler feed pumps, the emergency primary make-up pumps, and a 25 kW emergency turbo-generator set. In case of a total failure of all electrical power, these auxiliaries could be kept in operation either from steam produced by the reactor decay heat, or from the auxiliary boilers, which, even in the event of loss of their own auxiliaries, are designed to operate at 20 000 lb/h under natural draught.

Feedwater System

Both primary and secondary sides are fed from the same source of pure water, which is contained in three condensate tanks, each holding 500 U.S. tons.

Raw water is received from a 1 500 000-gallon tank fed from two sources, of which the Croton water contains considerably more dissolved solids than that which comes from the Croton watershed, and the plant is, therefore, designed so that it can deal with 100% Croton water. Treatment consists of a combination of cation exchange decarbonation, evaporation and further ion-exchange treatment to a purity of less than 0·01 ppm silica and less than 0·05 ppm ionizable solids.

Before passing to the condensate storage tanks or direct to make-up, the water is adjusted to a pH value of 8·8-9·0 by the addition of cyclohexylamine, and hydrazine is added as an oxygen scavenger. Although the deaerated water contains less than 0·007 ppm of oxygen, a trace does get into the boilers and 0·1-0·2 ppm of hydrazine is injected just before the boiler feed pumps. This "zero" treatment was selected rather than the conventional phosphate and caustic treatment to eliminate the possibility of stress corrosion cracking.

Waste Disposal

A simplified block diagram of the liquid waste disposal system is shown in Fig. 22. All radioactive wastes, except certain acid wastes, and laundry wash-water, are received in four 75 000-gallon tanks, and stored for decay of short-lived isotopes, and then passed for gas stripping, evaporation and ion exchange, after which the effluent is passed to the waste distillate storage tanks. It may then be passed to the purification system. If the amount of water in the purification system is high, the distillate will be pumped into the condenser discharge tunnel into the Hudson River, or, if monitoring shows that activity level is higher than permissible, it can be returned for reprocessing.

Laundry waste is separately collected, monitored and, if suitable, directly discharged, otherwise it can be passed to the waste tanks.

Concentrates from the evaporation process are transferred to

Table 2. Estimated Yearly Processing of Liquid Waste

Source	Activity (μc/ml)	Quantity (gal/year)	
		Normal	Max.
Operational			
Containment sump (moisture from ventilating air, wash-down, and possible relief valve discharge)	10^{-3}	90 000	180 000
Resin sluice water	10^{-3}	2 000	6 000
Gas system drains in chemical systems building	10^{-2}	50 000	150 000
Secondary blow-down (in case of primary coolant leak)	—	—	10 000
Valve leakage in containment vessel	4	6 000	12 000
Primary relief valves (1 blow/year 2 gal/day leakage)	4	1 000	15 000
Fresh water cooling system drainage (if heat exchanger leaks)	10^{-5}	1 000	5 000
Shut-down and Start-up			
Boric acid dilution (normal 3 start-ups/year)	5×10^{-2}	430 000	570 000
Decay heat system drainage	0·4	7 000	14 000
Maintenance and Refuelling			
Neutron shield tank drainage (max. once every 2 years)	10^{-5}	1 000	22 000
Central decontamination drainage	10^{-2}	12 000	24 000
Primary coolant system decontamination (4 boiler loops/year)	—	—	180 000
Cask wash water (prepn. of casks for shipment)	10^{-2}	12 000	36 000
Spent fuel storage pool water	10^{-5}	90 000	570 000
Containment vessel equipment drainage	0·4	20 000	40 000
Contaminated laundry water	5×10^{-4}	30 000	90 000
"Hot" showers and wash basins	10^{-4}	4 000	12 000
Laboratory Wastes			
Primary coolant system samples	4	50 000	100 000
Laboratory drains (washing of equipment)	10^{-2}	90 000	260 000
Miscellaneous			
Chemical systems building floor and equipment drains	3×10^{-2}	2 000	10 000
Chemical systems building sump	10^{-5}	15 000	30 000
Containment vessel foundation drain sump	10^{-5}	60 000	480 000
Total		973 000	2 816 000

the sludge storage tank, and are then routed, together with other solid waste, such as contaminated resins, to the drumming station. Here, they are mixed with cement-sand mixture and set in 55-gallon drums, for off-site disposal.

The estimated amount of liquid waste to be treated in a year is shown in Table 2.

Gas Handling System

Three gas handling systems are associated with the plant. The hydrogen blanketing system for purified water has already been briefly referred to. In addition, there are the sweep gas system and the waste gas system. The sweep gas system continuously circulates gas through various waste and drain tanks which, on account of evolution of active gas, cannot be left open to the air, and maintains a constant gas pressure in the tanks when the level of the liquid varies. It also accommodates the discharge from the primary vent system. Catalytic recombination is provided for hydrogen present; the water thus produced is routed to the sump tank of the liquid waste disposal system.

When the amount of gas in the system becomes excessive, a part will be discharged to the superheater stack through an absolute filter. If the activity level is too high for this, it is routed to hold-up tanks in the waste gas system.

Over-pressure protection is given by relief valves discharging to the stack at 15 psi. If radiation levels are too high, a valve automatically closes in the discharge line to the stack and, at 20 psi, emergency discharge is made to the sphere, through a 23 ft water seal.

The waste gas system is shown in simplified form in Fig. 23. Catalytic recombination is used for removing the hydrogen, which is reacted with measured amounts of oxygen to form water; a nitrogen blanket atmosphere is used to prevent formation of explosive mixtures. Other gases are stored in the hold-up tanks for 90-120 days, after which the estimated activity is not more than 3·2 μc/ml.

Laboratory analysis will determine the allowable discharge rate, so that when diluted with 300 000 ft^3/min of exhaust gas from the superheater and with a dilution factor of 3 000 from the superheater plume to the ground, maximum mean concentrations of Kr^{85} will not exceed those laid down by the AEC. Since very short release times would be required, the most advantageous meteorological conditions could be chosen for discharge.

ACKNOWLEDGMENTS

It has already been made clear that Consolidated Edison has itself undertaken the construction work, with the Vitro Corporation as nuclear consultants, while The Babcock and Wilcox Company were responsible for the reactor design and for the construction of the reactor and boilers. Other major contractors were:—

Chicago Bridge and Iron Company (containment sphere); **Foster Wheeler Corporation** (superheaters); **General Electric Company** (control rod drives and nuclear instrumentation); **Ingersoll-Rand Company** (condensers); **Westinghouse Electric Corporation** (turbo-alternator; safety monitoring equipment).

Our thanks are due to Consolidated Edison and The Babcock and Wilcox Company for making available the information upon which this article is based, and for affording site facilities to our artist for the pull-out drawing.

Rim Reinforced Openings—Part II

In Part I (*Nuclear Engineering*, Aug., 1961, p. 334) a description was given of the methods of calculating stresses near rim reinforced openings in a vessel subjected to pressure. Methods of dealing with end thrusts on a branch were also given. These methods are now being extended, so that stresses may be estimated around a branch which is subjected to a bending moment. Results from these two methods are also compared with those given when a bending moment is applied to an attachment rigidly connected to an infinite flat plate. The notation is given in Part I.

by **R. Kitching** and **Mrs. J. Perkins**
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MERIDIONAL MOMENTS (j_θ, j_θ^1, j_θ^{11})

Beam on Elastic Foundation Method

For a cylindrical shell, as in the analysis for end thrusts, the branch and shell are assumed split into strips 1 and 2 which are obtained by taking vertical sections through the axis of the branch. The interacting loads and moments at the junction of the strips, together with the radius of curvature for the strip of the cylindrical shell, will vary gradually with the angle θ (Fig. 1).

It is assumed that the junction of the branch and the cylinder is sufficiently rigid for the applied moment $\overline{M}_x$ to produce rotation through a small angle α at the root of the branch (Fig. 2). The radial deflection ω_θ of the cylinder at the joint will now be

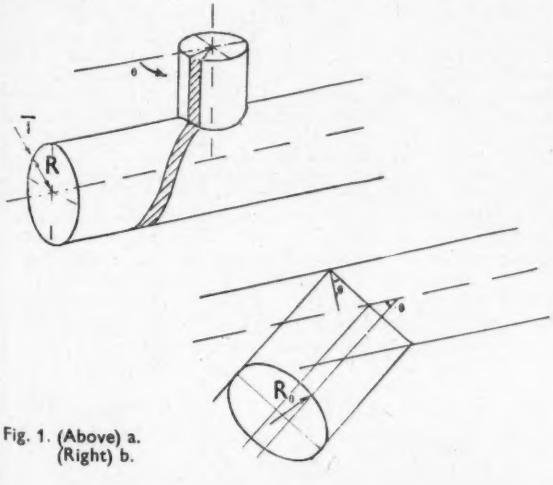

Fig. 1. (Above) a. (Right) b.

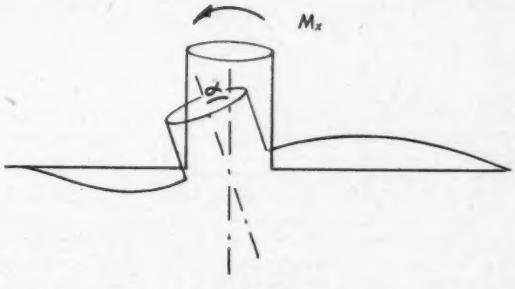

Fig. 2.

dependent on θ, and the boundary conditions for the strips 1 and 2 (Fig. 3) will be given by the following equations[1].

Radial deflection and rotation of the shell at the joint:

$$\omega_\theta = -(\beta_\theta M_\theta - P_\theta)/2\beta_\theta^3 D_1 = \alpha r \cos\theta$$
$$d\omega_\theta/dx = (2\beta_\theta M_\theta - P_\theta)/2\beta_\theta^2 D_1 = \alpha \cos\theta.$$

Radial deflection and rotation of a flush branch at the joint:

$$\omega_\theta^1 = -(\beta_2 M_\theta + Q_\theta)/2\beta_2^3 D_2$$
$$d\omega_\theta^1/dx^1 = (2\beta_2 M_\theta + Q_\theta)/2\beta_2^2 D_2 = -\alpha \cos\theta,$$

where

$$\beta_\theta = n/(R_\theta T)^{1/2} \quad \beta_2 = n/(rt)^{1/2}$$
$$D_1 = ET^3/4n^4 \quad D_2 = Et^3/4n^4$$
$$n^4 = 3(1-\nu^2)$$
$$\nu = \text{Poisson's ratio}$$

and $R_\theta = R \sec^2\theta$ is the appropriate radius of curvature of a strip of the shell cut at an angle θ.

From these equations, the end thrusts, radial load and moment per unit length of periphery can be found in terms of θ and α.

$$P_\theta = 2D_1\alpha(\beta_\theta^2 \cos\theta + 2\beta_\theta^3 r \cos\theta)$$
$$Q_\theta = -2\beta_2\alpha(\beta_2 D_2 \cos\theta + 2\beta_\theta D_1 \cos\theta + 2\beta_\theta^2 D_1 r \cos\theta)$$
$$M_\theta = 2D_1\alpha(\beta_\theta \cos\theta + \beta_\theta^2 r \cos\theta).$$

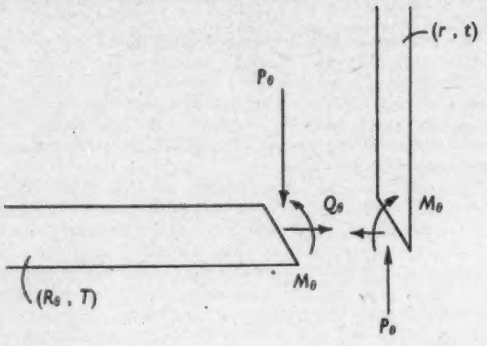

Fig. 3.

A shearing force N_θ will also exist at the junction of the branch and the shell, but its effect in balancing the applied moment is negligible.

For the equilibrium of the branch under the applied moment $\overline{M}_x$

$$\overline{M}_x/4 = \int_o^{\pi/2} M_\theta \cos\theta \,.\, r\,d\theta + \int_o^{\pi/2} P_\theta\, r \cos\theta \,.\, r\,d\theta$$

giving

$$\alpha = \overline{M}_x/8D_1\, r[2n/3(RT)^{1/2} + 3\pi n^2\, r/8RT + 16n^3\, r/15RT(RT)^{1/2}].$$

This gives rise to the following stress values, all of which are a maximum at $\theta = 0$.

$$\sigma_\theta = -Er\alpha/R$$
$$\sigma_r = \pm 6M_\theta/T^2 + Q_\theta/T$$
$$\sigma_H = E(\beta_2\, M_\theta + Q_\theta)/2\beta_2^2\, D_2\, r.$$

For a through branch, strip 2 is considered as an infinite beam, and the boundary conditions become[1, 2]:

$$\omega_\theta = -(\beta_\theta\, M_\theta - P_\theta)/2\beta_\theta^3\, D_1 = \alpha r \cos\theta$$
$$d\omega_\theta/dx = (2\beta_\theta\, M_\theta - P_\theta)/2\beta_\theta^2\, D_1$$
$$= -d\omega_\theta^1/dx^1 = -M_\theta/4\beta_2\, D_2$$

From these, the end thrust and moment per unit length of periphery are given by:

$$P_\theta = 2\beta_\theta^2\, D_1\, \alpha r \cos\theta \,.\, (4q_\theta + 1)/(2q_\theta + 1)$$
$$M_\theta = 4\beta_\theta^2\, D_1\, \alpha r \cos\theta \,.\, q_\theta/(2q_\theta + 1)$$

where

$$q_\theta = \beta_2\, D_2/\beta_\theta\, D_1 = (Rt^3/rT^3)^{1/2} \sec\theta = q \sec\theta.$$

From the equilibrium of the branch:

$$\alpha = \overline{M}_x\, RT/8n^2\, r^2\, D_1\left\{2K_1 + nrK_2(RT)^{-1/2}\right\}$$

where

$$K_1 = 2q/3 - \pi q^2/2 + 4q^3 - 4\pi q^4 + 16q^5 I$$
$$K_2 = 8/15 + 3\pi q/8 - 8q^2/3 + 2\pi q^3 - 16q^4 + 16\pi q^5 - 64q^6 I$$

and

$$I = (1-4q^2)^{-1/2} \log_e \left\{1 + 2q + (1-4q^2)^{1/2}\right\} \Big/ \left\{1 + 2q - (1-4q^2)^{1/2}\right\}$$

for $\frac{1}{2} > q > 0$,

or

$$I = 2(4q^2-1)^{-1/2} \tan^{-1}\left\{(2q-1)/(2q+1)\right\}^{1/2}$$

for $q > \frac{1}{2}$.

The stresses in the shell, which are a maximum at $\theta = 0$ are given by:

$$\sigma_\theta = -Er\alpha/R$$
$$\sigma_r = \pm 6M_\theta/T^2.$$

For a spherical shell, the analysis is simplified by the axial symmetry of the case, and the following solutions hold for a moment $\overline{M}$ applied in any direction.

For a flush branch

$$P_\theta = 2\beta_1^2\, D_1\, \alpha(1 + 2\beta_1\, r)\cos\theta$$
$$M_\theta = 2\beta_1\, D_1\, \alpha(1 + \beta_1\, r)\cos\theta$$
$$Q_\theta = -2\beta_2\, \alpha(\beta_2\, D_2 + 2\beta_1\, D_1 + 2\beta_1^2\, D_1\, r)\cos\theta$$

where

$$\beta_1 = n/(RT)^{1/2}$$

giving

$$\alpha = \overline{M}(RT)^{1/2}/2\pi D_1\, nr\left\{1 + 2nr/(RT)^{1/2} + 2n^2\, r^2/RT\right\}.$$

The stresses, which are a maximum at $\theta = 0$, are given by the same equations that apply to a flush branch on a cylindrical shell.

For a through branch

$$P_\theta = 2\beta_1^2\, D_1\, \alpha r \cos\theta \,.\, (4q + 1)/(2q + 1),$$
$$M_\theta = 4\beta_1^2\, D_1\, \alpha r \cos\theta \,.\, q/(2q + 1),$$

giving

$$\alpha = \overline{M}\, RT/2\pi D_1\, n^2\, r^2\left\{2q + nr(1+4q)/(RT)^{1/2}\right\}$$

and the stresses, again a maximum at $\theta = 0$, are given by the equations for the cylindrical shell.

Local Loading

The analyses for the cylinder and sphere by Bijlaard are again based on the assumptions that the loading is applied over some area of unpierced shell equivalent to the branch.

For the cylindrical shell the moment is represented by a double-triangular direct loading distribution (Fig. 4) acting over a square area whose sides $2c$ are such that its moment of inertia is equivalent to that of the branch. The solutions obtained by Bijlaard for bending moments M_x and M_ϕ, and membrane forces N_x and N_ϕ are given graphically[3] in terms of $\beta = c/R$ and $\gamma = R/T$. Fig. 5 is the graph for the bending moment in the longitudinal direction at the point X.

For the spherical shell an exact solution was found for a moment

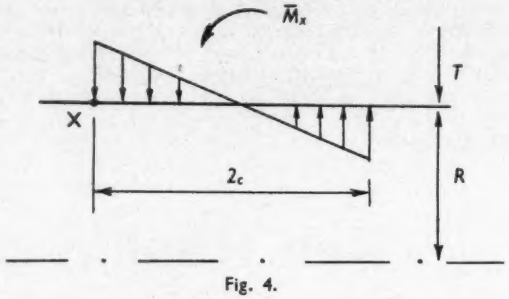

Fig. 4.

acting on a rigid circular attachment of radius r_o. The bending moments and membrane forces depend on the parameters $u = 1{\cdot}82(r_o/R)(R/T)^{1/2}$ and $s = ur/r_o$ and are shown graphically[4] as in Fig. 6.

An analysis allowing for the flexibility of a flush branch attached to a spherical shell has been carried out by Bijlaard[5] and, as before, the results are given graphically in terms of u and s for a range of the values of $\varrho = T/t$ and $\gamma = r/t$.

The stresses in the shell are given by the equation:

$$\sigma = N/T \pm 6M/T^2.$$

Infinite Flat Plate Analysis

This is particularly applicable to stiff branches where their

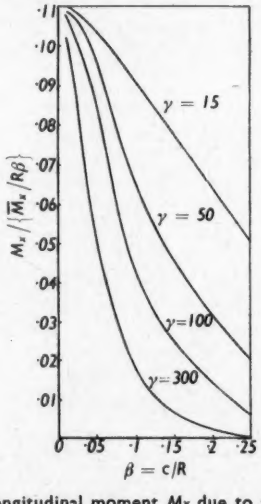

Fig. 5.—Longitudinal moment M_x due to an external longitudinal moment $\overline{M}_x$ applied to a cylindrical shell o $\gamma = R/T$.

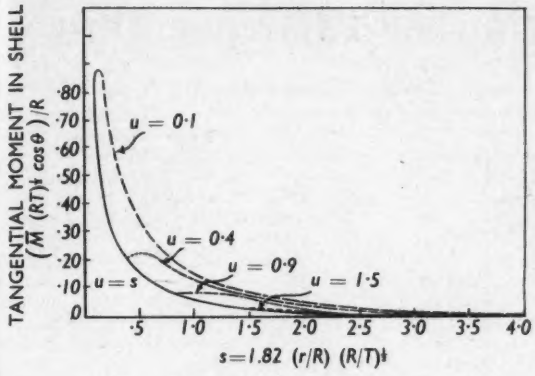

Fig. 6.—Tangential moments in a spherical shell due to an external moment $\overline{M}$.

diameters are small in relation to the shell diameter. The circular flat plate is assumed infinite and a moment $\overline{M}$ is taken to rotate the branch, assumed to be a stiff rim, through a small angle α. No account is taken of the flexibility of the branch.

The general equation for the deflection of a flat plate is :

$$\left(\frac{\partial^2}{\partial r^2}+\frac{1}{r}\frac{\partial}{\partial r}+\frac{1}{r^2}\frac{\partial^2}{\partial\theta^2}\right)\omega = 0.$$

It is assumed that

$$\omega = W\cos\theta.$$

The solution then becomes

$$W = A/r+Br+Cr^3+Fr\log_e r.$$

Substituting the following boundary conditions :

at $r = \infty$ $W = 0$,
$dW/dr = 0$,

at $r = a$ = radius of the hole $W = a\alpha$,
$dW/dr = \alpha$,

the constants may be determined.

By integration of the edge moments and equating to the applied moment, α may be determined in terms of $\overline{M}$, and the relevant stresses are given by the equations below.

$$\alpha = -\overline{M}/4\pi D \text{ where } D = ET^3/12(1-\nu^2).$$

Bending moment at the rim per unit length of circumference:

$$G = \overline{M}\cos\theta\,./2\pi a.$$

Shear force at the rim per unit length of circumference :

$$N = \overline{M}\cos\theta\,./2\pi a^2.$$

The stresses are given by

$$\sigma_r = N/T \pm 6G/T^2,$$
$$\sigma_\theta = \nu\sigma_r.$$

Comparison of Methods

For the 50-ft diameter cylindrical vessel considered in Part I, which has 4-in thick walls and a branch tube of 54-in bore with 2¼-in thick walls, the circumferential and radial stresses in the shell due to a bending moment of 100 ton/in applied to the branch have been calculated by the different methods and the results are shown in the accompanying table.

CIRCUMFERENTIAL MOMENTS (j_5, $j_5{}^1$, $j_5{}^{11}$)

This, of course, only applies to the cylinder. It is not possible to obtain a solution to this problem by the beam on elastic foundation method, as the assumptions for an elastic foundation are not valid due to the curvature of the cylinder in this plane.

The solution by the local loading analysis method is similar to that given for a moment applied in the longitudinal direction[3].

Adjacent Openings

High stresses in a shell in the locality of a rim usually decrease rapidly with increasing distance from the centre of the branch. The rate of decrease depends greatly[6] on the quantity $(RT)^{1/2}$.

RIM REINFORCED OPENINGS

Method	Circumferential stress, ton/in²	Radial stress, ton/in²
Elastic foundation—flush branch	−0·0627	±0·2233–0·0573 = −0·2806 or +0·1660
Elastic foundation—through branch	−0·1473	±0·1630
Local loading	±0·104–0·031 = −0·135 or +0·075	±0·173–0·008 = −0·181 or +0·165
Infinite flat plate	−0·0626 or +0·0598	±0·204–0·0047 = −0·2087 or +0·1993

As a general rule it can be assumed that no significant interaction between the forces at a branch and those at an adjacent discontinuity (e.g., another branch) will take place if the distance x measured along the shell from the rim is greater than that given by $x = 2(RT)^{1/2}$.

Maximum Stress Related to Design Stress

In design, the total stress concentration, which is the sum of the individual stress concentrations, is generally taken as satisfactory if it is below yield. It is recommended, for calculation of the overall factor, that, where alternatives are given, the method giving the most pessimistic stress value is used, although due allowance must be given for the actual case considered; e.g., for a relatively stiff branch, the local loading method would probably be nearest the truth.

The rules laid down in B.S. 1500 : 1958 for reinforcement of openings are a good guide, although it has been shown by experiment[7, 8] that symmetrical reinforcement is far superior to reinforcement on one side of the shell only. There is no doubt, for the type of reinforcement considered here, that a through branch, coupled with, if necessary, a symmetrical thickening of the parent shell, is the most desirable form provided the weld detail is a satisfactory one.

It is not only unnecessary, but usually undesirable to have a replacement factor which exceeds about 1·4, as radial stresses in the shell (from the geometric effect) become dominant (see Figs. 5a and 5b of Part I) and high tensile stresses occur in an undesirable direction from the point of view of the welded joint between shell and branch.

Factors used in the codes to obtain design stresses were originally factors of ignorance, allowing for the high stresses occurring at discontinuities such as branch connections. For this reason most of the plate used for a large pressure vessel operates at much lower levels of stress than it is capable of withstanding. Now that the stress levels in the regions of the discontinuities can be reasonably assessed there is no longer any need for this state of affairs. As suggested in the recent Institute of Mechanical Engineers Symposium[9] the code design stresses used for the majority of the shell could be raised, and the local discontinuity stresses dealt with separately. For large nuclear vessels this would certainly lead to economies and would involve no more detailed stress analysis in the design office than is at present carried out for the satisfaction of designer and customer.

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Measuring the Concentration of Charcoal Dust

by H. F. Enxmann
(*ANEA Dragon Project*)

CHARCOAL is used extensively in the OEEC Dragon reactor experiment as a coolant clean-up medium. In some circumstances—during the initial filling of the beds and subsequently during actual operation—the coolant may become contaminated with charcoal dust. This has led to the development of a system for continuously measuring the concentration of dust. The method is based on the inflammability of the charcoal dust. The measurement of its concentration is done indirectly by gas analysis.

The dust laden air, leaving the charcoal bed, the elutriation column, the cyclone or the filter device under investigation is passed through a furnace, where the charcoal dust is burnt. The dust is thereby converted into CO_2. The exhaust from the furnace is continuously sampled and its CO_2 concentration measured and recorded by means of a CO_2 infra-red gas analyser and recorder. From the flow rate and the CO_2 concentration the rate of dust carried in the air stream and the amount of released charcoal dust can be calculated.

Figure 1 shows the flow sheet of the set-up as it was used in this experiment. Compressed air was led through a combined CO_2 trap and through a dryer to remove the normal CO_2 content and the moisture of the air which would cause too high a background reading. The flow rate was measured by a rotameter. The air stream was then passed through the charcoal bed. Provision had been made to enable the by-passing of the bed for regulating the flow-rate through the bed. The dust laden air was then passed through a furnace packed with crushed alumina and externally heated. The solid bed filling gave a reliable transfer of the heat to the dust particles, necessary to initiate the combustion.

For the removal of the initial CO_2 content the best combination was found to be a trap where the sodalime was followed by an equal volume of silica gel. Without the silica gel and also in the case of placing the silica gel in front of the sodalime the CO_2 infra-red gas analyser gave an indication of 50 v.p.m. This was due to an excessive water vapour content, which influenced the reading of the infra-red gas analyser.

So far two different problems have been investigated, using this method:

(a) The removal of the initial dust content in charcoal samples by wind sifting.

(b) The dust generation in charcoal beds at different flow rates of air.

(a) In the first case samples of 10 cc charcoal were thrown into the glass test cylinder against an upwards flow of processed air at different velocities (6 to 29 cm/s according to the cross-section of the empty tube). The evaluation of the recording

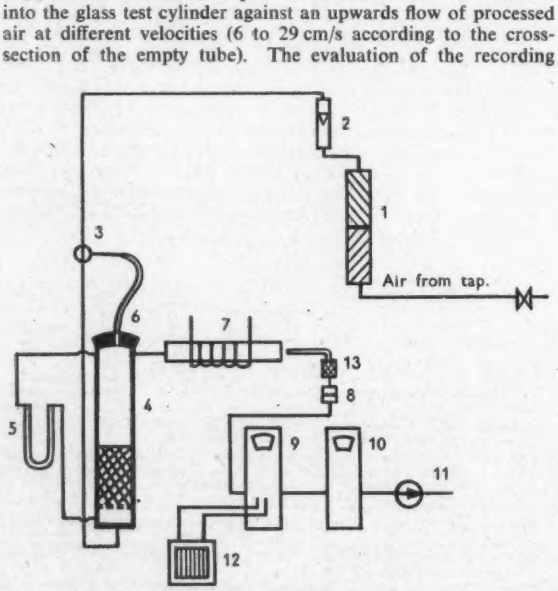

Fig. 1.—Flow diagram of experiment.

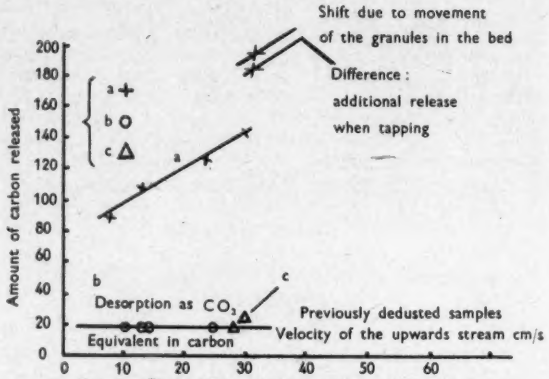

Fig. 2.—Elutriation curves and desorption tests.

indicated, for increasing flow rates, increasing amounts of CO_2 from which the contribution due to desorption of CO_2 from the charcoal sample had to be deducted. This was within an accuracy independent of the flow rate. The results from desorption alone are shown by curve (b) in Fig. 2. The amount of dust released in this way from the thrown-in charcoal sample as a function of the upwards velocity of the air is shown by curve (a) in Fig. 2. The amount of dust registered for the flow velocity of 29 cm/s was considerably larger than for the lower velocities, as with this flow rate the charcoal granules, after falling down the elutriation cylinder, did not immediately come to rest, but arranged themselves within the bed to a preferred position, where there was the minimum drag caused by the air flow. This arranging movement tended to rub off the dust particles which were still attached to the granules. although according to the air stream velocity, this dust should have been entrained before.

(b) The dust generation in a charcoal bed, while passing air through it at flow rates low enough not to move the charcoal granules, was too small to be detected (less than 3 μg/min) with this first experimental set-up. Only when the flow rate passing through a 70 cc bed in a 1½ in bore tube was increased to 46 cm/s (referring to the cross-section of the empty tube), was an initial reading of approximately 5 μg/min obtained, which after about 10 s became too small to read. By that time the granules of the uppermost layer of the bed came to rest in positions where there was the minimum of drag forces exerted on them. This arranging movement always started somewhere near the wall of the cylinder, followed initially along the periphery, moved then to the axis to finish up somewhere in between, clearly illustrating the velocity profile in the charcoal bed with its flow channelling along the walls. With the flow rate increased to 55 cm/s the dust concentration in the air stream rose initially to 27 μg/min while the upper half of the bed carried out a vigorous pulsating movement, which after about half a minute was followed by a strong bouncing movement of the upper third of the bed with a 10 μg/min recording and finally yielded a constant value of approximately 5 μg/min still in connection with a slight movement amongst the granules at the top of the bed. It was hoped that a connection between the pressure drop across the bed and the rate of dust generated per unit bed cross-section could be found. But it showed that measurable dust generation was only produced when the inversion point of the pressure drop curve was passed, which is the case when the bed starts to fluidize. In order to see the influence of the length of the charcoal bed on the rate of generated dust, and especially to investigate the filtering effect of the charcoal on its own dust, an attempt was made to repeat the same measurements with a bed four times longer. But the longer intervals required to reach equilibrium from this larger charcoal bed gave readings which were no longer reliable.

Trapping Fission Products in the HTGR†

by R. M. WATKINS, D. D. BUSCH and L. R. ZUMWALT
(*General Atomic*)

DURING the early months of 1960, experimental apparatus was constructed at General Atomic for duplicating as closely as possible the operating conditions anticipated for the fuel element purge gas system of the High-Temperature Gas-cooled Reactor (HTGR). Employing this apparatus, tests of the iodine and tellurium trapping capabilities of silver and copper-coated charcoal at temperatures up to 900°F were conducted.

It was observed that metal-coated charcoal exhibits much better holdup characteristics compared to metal mesh or turnings alone. Very sharp boundaries were noted in the iodine and tellurium vapour-helium chromatograms of these reagents, and it appeared that the migration of silver and copper iodides through the trapping bed is suppressed significantly by the presence of charcoal. These tests were conducted employing a helium carrier gas flow rate equal to that expected under reactor operating conditions (based on the passage of 0·25 lb/h of helium through each fuel element). In order to obtain results more rapidly, the iodine and tellurium concentrations were adjusted to be approximately 300 times those which should exist during actual HTGR operation.

In order to evaluate more fully the use of copper and silver as trapping agents, it is necessary to run tests at partial pressures of iodine similar to those expected in the actual fuel element purge system (approximately 2×10^{-6} atm). This need arises because the vapour pressure of AgI and CuI at trap temperature is of the same order of magnitude as the partial pressure of iodine. Under these conditions the role of the charcoal is of critical importance. Refrigeration and vacuum apparatus needed for this study was constructed in the summer of 1960, and tests making full use of this equipment were started in the third quarter of the year. By late October it had become obvious that there was a significant difference between the iodine delay characteristics exhibited by different varieties of charcoals and that the effect of temperature on the breakthrough times was most pronounced.

Soon after the Dragon Reactor–HTGR–EGCR information exchange meetings, which were held in San Diego and Oak Ridge in early November 1960, a new system for rating the trap materials under test was adopted. The capacity for delaying fission products is now measured in terms of Reactor Equivalent Delay (in days). These RED values give an indication of how long a particular fission product atom entering one of the high temperature traps can be expected to remain (neglecting the decay factor). In the case of I^{131}, the most abundant iodine fission product, the half life is only 8·1 days. Thus, a delay of 10 half lives or 81 days would result in a decrease in iodine activity by a factor of 2^{10} or 1 024. This long a delay (equivalent to a RED value of 81) would therefore reduce the iodine concentration leaving the trap to less than 0·1% of its original value.

By the end of 1960, 35 long-term radioiodine adsorption tests had been completed. In all these tests an attempt was made to duplicate HTGR operating conditions as closely as possible. An average flow rate of 20·7 cm³/min (measured at 25°C and 1 atm) was maintained and an iodine partial pressure of approximately 5×10^{-6} atm was employed. As the exact operating temperature of the internal trap will depend on the precise trap design which is not yet fixed, tests continued to be run at several different temperatures. The results of these studies for uncoated charcoals are presented in Table 1, while the results of the metallic-coated reagent evaluation tests are shown in Table 2. It can be seen from the first table that coconut charcoals exhibit greater delay characteristics for iodine than do charcoals of coke origin. This is apparently due to the very fine grain structure of coconut shells which permits the development of a maximum number of micro-pores when they are converted into activated charcoal. The resultant greatly increased internal surface area makes this material ideally suited for application in gas purification work.

TABLE 1
Reactor Equivalent Delay for Iodine (in days)*

Trapping Reagent				Test temperature	
Charcoal designation	Type	Mesh size	Manufacturer	850°F	1 000°F
BPL ..	Coke	12×28	Pittsburgh Chemical	6	—
HCC ..	Coke	12×28	National Carbon ..	16	5
L-5578-1 ..	Coconut	12×20	Barnebey–Cheney ..	—	46
MI-1 ..	Coconut	12×20	Barnebey–Cheney ..	137	51
PCB ..	Coconut	12×28	Pittsburgh Chemical	143	59

TABLE 2
Reactor Equivalent Delay for Iodine (in days)*

Trapping reagent			Test temperature		
Charcoal	Mesh size	Metallic coating	850°F	1 000°F	1 150°F
HCC ..	12×28	37 wt-% Cu ..	—	13	—
BPL ..	12×28	16 wt-% Ag ..	>200	59	—
HCC ..	12×28	37 wt-% Ag ..	>200	63	8
HCC ..	12×28	15 wt-% Ag ..	>200	70	—
1/4 HCC over 3/4 MI-1 ..	12×28	37 wt-% Ag ..	>200	70	—
	12×20	Uncoated			

*RED values calculated on the basis of 24 channels per trap, each channel 11 inches in length and ¼ in in diameter. Helium flow rate = 20·7 cc/min (equivalent to 0·25 lb/h per HTGR fuel element).

The strong dependence of delay time on temperature is clearly shown in Table 2. It appears that no matter what the incoming iodine concentration, if there is excess unreacted metal in the trap, the exit iodine concentration (neglecting decay) will depend only on the vapour pressure and stability of the metal iodide formed. Thus copper, which forms a more volatile iodide than silver, offers less delay for iodine, whereas various densities of silver coating offer similar delay values since in all cases a considerable excess of silver metal is present. The methods employed in applying the metal coatings and the base materials used result in some variation in RED values, but by far the greatest effect is due to the vapour pressure and stability of the iodide in the trap.

Early in 1961 an attempt was made to employ tagged strontium in trapping studies. Sr^{85} in the form of the nitrate was received from Oak Ridge National Laboratory. After conversion to the carbonate and then to the oxide, the metal was formed directly in the test apparatus via reduction with zirconium turnings. After several runs employing this procedure, work along this line was terminated as we were unable to detect any vapour phase transport of strontium metal into the test trap. On the basis of these results it appears unlikely that strontium transport will present a significant problem in the operation of the HTGR internal trap.

More positive results have recently been achieved in tests of tellurium trapping employing Te^{121} which was produced via neutron irradiation of enriched Te^{120} in the Oak Ridge Research Reactor. Both the 37 wt-% silver-coated charcoal and a new caesium impregnated charcoal reagent were employed in these tests. The results, as well as those of a study employing a mixture of tagged iodine and tellurium are presented in Table 3. It is

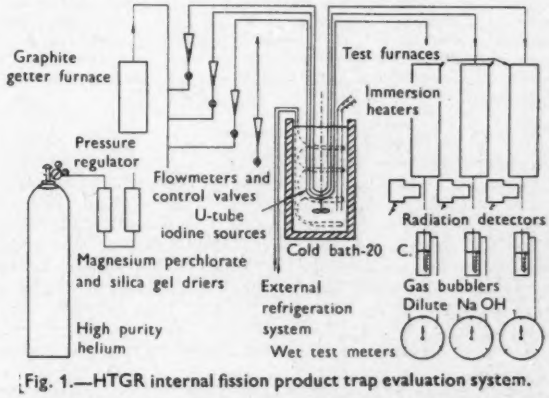

Fig. 1.—HTGR internal fission product trap evaluation system.

† From a paper "High Temperature Fission Product Trap Study for Gas-cooled Reactors," presented at a meeting of the American Nuclear Society, Pittsburgh, June, 1961.

TABLE 3
Summary of Recent Test Results

Material under test	Nuclide	Concentration	Trap temp.	RED value* (days)
37 wt-% Ag coated coke base charcoal†	Te^{131}	5×10^{-6} atm	1 000°F	>600
37 wt-% Ag coated coke base charcoal†	I^{131} Te^{131}	5×10^{-6} atm 3×10^{-5} atm	1 000°F	>130
Cs impregnated (1/4 g/g) coconut charcoal‡	Te^{131}	3×10^{-5} atm	1 000°F	>460
Cs impregnated (1/4 g/g) coconut charcoal‡	I^{131}	5×10^{-6} atm	1 000°F	115

* RED values calculated on the basis of 24 channels per trap, each channel being 11 in long and 1/4 in in diameter. Helium flow rate = 20·7 cc/min (equivalent to 0·25 lb/h per HTGR fuel element).
† National Carbon HCC 12×28 mesh.
‡ Barnebey-Cheney MI-1 12×20 mesh.

interesting to note that after more than 600 equivalent days of reactor operation at 1 000°F, no tellurium had been detected downstream of the silver charcoal trap. The presence of tellurium even appears to suppress the iodine breakthrough time. This may be due to the decrease in iodine volatility resulting from the formation of tellurium iodide prior to entering the trapping zone.

Based on the results presented in Table 3, it appears that there will be little difficulty in removing all significant tellurium activity from the purge stream at 1 000°F.

Preparation of Trapping Reagent

An extensive series of experiments has been conducted on improved procedures for producing copper- and silver-impregnated charcoal for fission-product trapping. Prolonged soaking of charcoal in aqueous, ammoniacal and alcoholic solutions, employment of wetting agents, adsorption under vacuum, high-speed centrifuging, and techniques involving the reduction and decomposition of metallic salts and complexes were evaluated. On the basis of the results of these experiments, a process has been developed which will produce charcoal containing up to 50 wt-% copper or silver.

The present method which is employed for the preparation of silver-charcoal trapping reagent takes advantage of the ability of activated charcoal to reduce silver salts in solution. To slow this process and achieve a tightly adherent coating of the metal, ammonia is employed to complex most of the available silver ions and thus reduce the reaction rate. Silver acetate has been selected as the source of the silver and it is employed in a saturated ammoniacal solution. Table 4 shows the improved coating quality of the ammoniacal over the aqueous silver acetate solution.

The apparatus employed in HTGR internal fission product trapping studies is shown schematically in Fig. 1. It consists of a source of purified helium, several drying columns (employing magnesium perchlorate and silica gel), a vertically mounted getter furnace containing graphite (maintained at 1 800°F), a distribution manifold with four flow meters, a shielded metal dewar flask with built-in refrigeration unit (capable of maintaining −40°F),

TABLE 4
Silver-Charcoal Coating Characteristics

Coating reagent	Charcoal	Mesh size	Wt.-% silver
Aqueous silver acetate .. (saturated solution)	National Carbon Co. HCC ..	12×28	6·5
	Pittsburgh Chemical Co. BPL	12×28	10·6
Ammoniacal silver acetate (saturated solution)	National Carbon Co. HCC ..	12×28	30·2
	Pittsburgh Chemical Co. BPL	12×28	31·4

three 2-ft long, 1-in diameter tube furnaces (capable of maintaining 1 850°F), three gas bubblers (containing sodium hydroxide solution) for final gas cleanup, and three wet test meters for measuring total gas flow volumes.

Tagged iodine is injected into the helium stream at the desired partial pressure from "U" tubes located in the metal dewar flask, while tagged tellurium may be injected into the helium in the heated zone directly above the test traps.

The trapping reagent under study is contained in 30-in long, ¼-in diameter quartz furnace tubes. The test trap usually consists of a ½-in deep bed of charcoal reagent held in place by Fiberfrax plugs. A temperature control thermocouple is attached to the outside of the furnace tube directly opposite the test trap, while other thermocouples are employed to check the temperature of the iodine and tellurium sources. A second series of charcoal traps is located in the cooler zones at the bottom of the test furnaces. These are continuously monitored for activity buildup.

Trap evaluation with a really complete spectrum of fission products has been delayed until startup of the HTGR in-pile loop test.

Removing Radioactive Iodine from Gaseous Effluent

by J. F. W. SMITH and
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(APD, The English Electric Co., Ltd.)

In a previous paper by Geldart[1] the problems involved in removing iodine from reactor circuit gas streams have been discussed and various adsorbents evaluated. In this paper the particular problems involved in civil reactors of the Calder type are dealt with in rather more detail and experiments described in which tests were carried out with carbon, copper, bronze and silver adsorbents.

THE release of iodine in a reactor may be fast or slow. A slow release is usually the result of a can failure and is not serious as it has been shown that iodine will be adsorbed on various reactor materials such as graphite, steel, magnesium and activated alumina[2]. There is the possibility of desorption at temperatures exceeding 350–450°C and it is unlikely in a slow release that the reactor gas will be seriously contaminated by iodine. A fast release such as would occur in a single channel fire, would, however, lead to a serious contamination of the reactor gas. This, therefore, constitutes a major problem. The conditions which give rise to this fast release may require immediate shut-down, and subsequent blow-down of the gas circuit.

Iodine adsorbed on submicron particles can be removed by filters or vented to atmosphere with little hazard but the same cannot be said of iodine vapour, which must be adsorbed on to some medium.

Adsorbents for Vapour Iodine

Liquid adsorbents can be ruled out for Calder-type reactors because of the high temperature of the gas (200–350°C), and this also rules out a number of the proposed solid adsorbers. Our studies have been confined to sintered bronze, copper mesh, silvered copper mesh and activated carbon. Previous work on some of these materials has already been published[3, 7, 8, 9]. These papers indicate that activated charcoal is superior to both copper and silver at room temperature. However, there is practically no data on adsorption in the range in which we are mainly interested (200–300°C). It was necessary to evaluate the adsorbents in terms of the following requirements:—

1. An efficiency of greater than 99% (99·9% or better is desirable).
2. A capacity sufficient to retain all the iodine expected to be

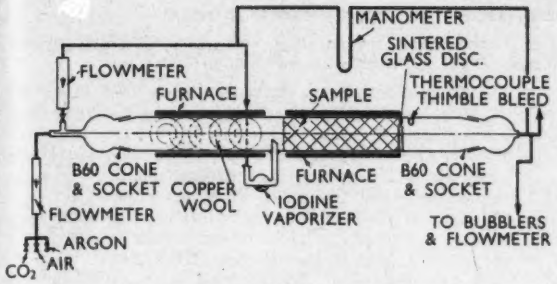

Fig. 1.—Large adsorption apparatus.

released as vapour during an incident such as a single channel fire.

3. Efficient in operation between temperatures of 20–350°C, at linear face velocities of 0·5–2·5 ft/s.
4. Satisfactory operation in the presence of carbon dioxide and/or air and water vapour.
5. Long service-free life.
6. Low resistance to gas flow, and occupy as small a volume as possible consistent with efficiency.
7. Easily retained in the containment vessel under conditions of high face velocities.
8. Fire-proof (up to 350°C in air).

Experimental Rig

The type of rig used is shown diagrammatically in Fig. 1. Two models were used; in the first the adsorbent bed was 3 in long and ½ in diameter, in the second the bed was 9 in long and 2 in diameter. The first bed was used for preliminary sorting purposes, the second for more detailed studies. The source of iodine for these experiments was potassium iodide to which I^{131} (supplied by The Radiochemical Centre, Amersham) in iodide form was added. The potassium iodide was converted to iodine and extracted into ethereal solution, and a known volume of this was placed in the iodine evaporator, after temperature equilibrium in the apparatus has been established at the required flow-rate. The ether evaporated rapidly, and the remaining iodine crystals were subsequently vaporized at a controlled rate, by passing a small fraction of the total gas stream through the evaporator. The furnace temperatures during a run were recorded and also the gas temperature at the point indicated; the temperature at the inlet end of the bed was measured during the pre-heat period by a thermocouple inserted via the iodine evaporator, but this was removed during the actual run, after temperature equilibrium had been established. The carbon dioxide was not filtered, but as it was obtained from liquid CO_2 it was assumed to contain very few condensation nuclei.

The efficiencies of the various beds were measured by counting the I^{131} using a liquid scintillation counter. The I^{131} collected in the sodium thiosulphate bubblers could be measured directly using an aliquot of the solutions; the I^{131} retained on the metal beds (bronze, copper and silver) could be removed by extraction with 2% potassium cyanide solution, and then measured after taking a suitable aliquot of the extract. Knowing the amount of I^{131} added initially a % recovery could be calculated, and the ratio of I^{131} on the bed to the I^{131} in the bubblers gives a measure of the efficiency of the bed. (In the case of the carbon beds, the efficiencies were measured by assuming a 100% recovery as it was not possible to remove completely the iodine from the carbon by extraction.)

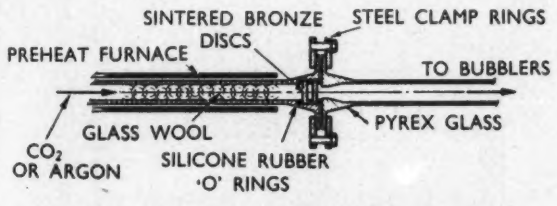

Fig. 2.—Apparatus for holding sintered bronze specimens.

Distribution of iodine along the bed was estimated using an end-window G.M. tube; the tube containing the adsorbent was moved along a lead alley which had in it a slot allowing 1 mm of adsorbent to be counted. In the case of the metal beds this distribution could be measured more accurately by removing the bed (after adsorption) in sections, and extracting each section separately.

The rig used for testing bronze discs is shown in Fig. 2. The experimental procedure was similar, but the maximum temperature was limited to 250°C because silicone rubber O rings were used to seal the discs into the apparatus.

Sintered Bronze Adsorbent

Tests with sintered bronze discs (supplied by Bound Brook Bearings Ltd.) of 1 mm thickness, showed that this material (if used in the particulate filters upstream from the main iodine adsorbent bed) would retain varying amounts of vapour iodine at expected temperatures (200–250°C) and flowrates. At 250°C and 2·5 ft/s linear flow velocity over 98% of the iodine is removed. At 1 ft/s (the anticipated flow through the particulate filter) and 200°C, over 99·4% of the iodine is removed and at 0·25 ft/s and 200°C 99·9% of the iodine is removed. These tests indicate that in an iodine filtration system employing sintered bronze particulate filters, these can be expected to remove a variable amount of iodine.

Copper Mesh Adsorbent

Tests with copper mesh (supplied by Hydronyl Sindicate Ltd.), using a 3 in bed at various temperatures, indicated that this was not an ideal material. At 250°C and 2·5 ft/s, 96·7% iodine was removed, at room temperature only 80% was removed, and above 300°C the efficiency falls off rapidly, being about 50% at 400°C. Tests in 50% air+50% CO_2 and in presence of water vapour (1%) indicated that under these conditions, efficiencies were much less. Iodine adsorbed by copper was not removed by heating in vacuum at 250°C for 15 h, but iodine adsorbed by copper at 250°C, was removed by heating to 400°C (cuprous iodide condensing in the cooler part of the apparatus).

TABLE 1

Retention of Iodine by Silvered Copper Mesh

Constants
(a) Material: silvered copper knitmesh.
(b) Gas flow 0·17 ft³/min=2·5 ft/s (calculated at temperature of experiment).
(c) Cross-sectional area of adsorbent (copper knitmesh) approximately 0·0011 ft²
(d) Length of adsorbent 0·25 ft.
(e) Weight of adsorbent—8 g.
(f) Approximately 89% voids.
(g) 5 mg of iodine per run.
(h) Differential pressure drop across the bed 3–5 cmHg.
(i) Duration of test approximately 1 h.

Temperature °C	% Iodine retained on adsorbent	Distribution of iodine on adsorbent		
		% on 1st cm	% on 2nd cm	% on remainder
30	92·2	38·8	22·6	36·6
90	98·8	51·8	29·2	19·0
140	98·1	58·8	22·6	18·6
200	99·6	65·2	22·0	12·2
255	99·7	68·2	17·6	14·2
330	99·3	85·1	12·1	2·3
395	99·2	85·6	3·2	11·9

Silvered Copper Mesh Adsorbent

This material was first prepared by simple dipping of clean copper mesh in a silver cyanide bath but was later obtained in a more stable form from Hydronyl Syndicate. This material shows considerable promise, as Table 1 indicates. In the temperature range in which we are principally interested (200–350°C) over 99% of the iodine is removed from the gas stream by a 3 in bed at 2·5 ft/s linear face velocity. (Efficiency drops off at room temperature, however, where the rate of reaction is much slower.) Tests in 50% air+50% CO_2 and in presence of water vapour (1%) indicated that under these conditions efficiencies were equally good provided that the silver coating was sufficiently thick to protect the copper from oxidation.

Using a 9 in bed, the silvered mesh was tested at various packing densities as shown in Table 2. 96% voids represents the normal density of the material with no compression while 93·2% represents considerable compression. It will be seen that the compression

makes little difference. The run at room temperature confirms the previous finding that the reaction is slow at room temperature.

Further tests carried out on 9 in beds with increased flow-rates are shown in Table 3. It will be seen that the efficiency falls off slightly, but that at 5 ft/s the difference is very small (compared with that at 2·5 ft/s).

The distribution of iodine on the bed confirms the findings of other workers, viz: iodine appears to be present in two forms, one of which, probably elemental iodine, is adsorbed rapidly and quantitatively while the other is adsorbed slowly, and non-quantitatively. This second form may be iodine adsorbed on condensation nuclei, but there is insufficient evidence to state this categorically.

TABLE 2

Effect of Compression on Iodine Adsorption Efficiencies

Constants
(a) Silvered copper mesh (ex. Hydronyl Syndicate Ltd.).
(b) Gas flow 3·5 ft³/min=2·5 ft/s (at temperature of experiment).
(c) Cross sectional area of adsorbent 0·018 ft².
(d) Length of bed 0·75 ft.
(e) 10 mg iodine per run.
(f) Duration of test approximately 100 minutes.

Temperature °C	% Iodine retained on adsorbent	Distribution of iodine on adsorbent % on 1st cm	% on 2nd cm	% on remainder	% Voids
300	99·50	98·3	1·5	0·2	93·2
295	99·45	85·9	11·9	2·9	96·0
30	97·20	30·7	24·8	44·5	96·0

Activated Carbon

In initial experiments with activated carbon from three sources, British Drug Houses (an activated charcoal for gas adsorption), Hopkin and Williams and British Carbo-Norit " Grade SCII," it was shown that on a 3 in bed at 2 ft/s linear face velocity 99·98% iodine is adsorbed in the temperature range 0–300°C. Unfortunately, the ignition temperatures of these grades in air were in the range 280–350°C[10] and they were therefore unacceptable for our purpose. (See requirement No. 8.) We therefore investigated two other grades of carbon, British Carbo-Norit CP (8–14 mesh) and Sutcliffe Speakman 207B, both of which had an ignition temperature of over 400°C. Table 4 shows the results of tests on these materials.

It will be seen that where a vertical bed with downward flow of gas was used, the efficiencies of both grades is excellent either in CO_2 or in CO_2/air mixtures. (The poor efficiency of the carbon in a horizontal bed was attributed to channelling, induced by the high gas flow.) Activated carbon (of high ignition temperature) appears therefore to be superior to silvered copper as an iodine adsorbent. This is perhaps due to its capacity for adsorbing iodine present in forms other than elemental iodine. It has, however, certain minor disadvantages, such as a relatively high pressure drop across the bed and a tendency to disintegrate mechanically at high flow rates.

TABLE 3

Effect of Increasing Gas Flows

Constants as for Table 2 (except for gas flow)

Temperature	Velocity ft/s	% Iodine retained on adsorbent	Distribution of iodine on filter % on 1st cm	% on 2nd cm	% on remainder
200	2·5	99·48	76·9	17·6	5·5
215	5·0	99·45	57·7	25·2	17·1
185	8·25	99·29	35·8	24·0	40·2
315	2·5	98·84	86·3	12·1	1·6
293	5·0	98·81	76·9	17·4	5·7
296	8·25	94·00	79·8	13·7	6·5
30	8·25	94·57	37·9	19·6	42·5

Composite Carbon/Silvered Copper as Adsorbent

It would appear that a composite activated carbon/silvered copper bed would have the following advantages:—

(a) The pressure drop across the bed would be lower than if the whole bed were of activated carbon (silvered copper mesh has negligible resistance to flow).
(b) Silvered copper (upstream) would retain the bulk of the molecular iodine, so that in the improbable event of a fire in the carbon bed, this iodine would not be released.
(c) The carbon would remove the iodine present in the form which is not readily adsorbed by the silvered copper.
(d) In the initial rush of gas during a " blow-down," while the beds were heating up from ambient temperature, the low efficiency of the silvered copper would be off-set by the greater efficiency of the carbon. By using a composite bed one would ensure good efficiency almost independent of temperature, up to 400°C.

Tests on composite beds (see Table 5) show that in fact they do have an excellent efficiency both in CO_2 and in CO_2/air mixture. An 8 in bed was tested, of which 3 in was silvered copper (upstream) and 5 in activated carbon (BCNU. CP 8/14 mesh).

A composite silvered-copper/activated carbon bed can therefore be recommended.

TABLE 4

Retention of Iodine by Activated Charcoals

Samples	Temperature °C	% Iodine retained in adsorbent	Gas flow ft/s	Wt. of adsorbent g
*S & S 207B	200	99·99	1·88	169
*BCNU.CP	200	99·96	1·88	208
*BCNU.CP	300	>99·98	2·00	170
*S & S 207B	300	99·98	2·00	178
*S & S 207B	25	99·85	2·00	180
*BCNU.CP	25	>99·95	2·00	168
†BCNU.CP	325	85·20	4·00	270
†S & S 207B	200	89·70	1·38	120

S & S 207B=Sutcliffe and Speakman grade 207B.
BCNU.CP =British Carbo-Norit Union grade CP.

* Vertical bed.
† Horizontal bed.

CONCLUSIONS

A filtration system for removal of iodine from contaminated gas circuits of a Calder-type reactor should consist of the following:—

(a) A particulate filter to remove uranium oxide particles (sintered bronze is recommended).
(b) An " absolute " type filter to remove sub-micron particles.
(c) An iodine vapour adsorbent (a composite bed of silvered copper mesh and high temperature activated carbon is recommended).

ACKNOWLEDGMENTS

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TABLE 5

Retention of Iodine on a Composite Bed and the Effect of Contaminant Air on the Efficiency

Samples	Temperature °C		% Iodine retained on adsorbent	Gas flow ft/s	Wt. of adsorbent	% I^{131} on Ag/Cu
* Mixed bed	200	CO_2	>99·99	1·88	49 g Ag/Cu 119 g Carbon	9·67
* Mixed bed	200	CO_2	99·95	1·88	49 g Ag/Cu 135 g Carbon	Not measured
* Mixed bed	300	CO_2	>99·95	2·00	49 g Ag/Cu 110 g Carbon	85·5
* Mixed bed	25	CO_2	>99·95	2·00	48 g Ag/Cu 122 g Carbon	Not measured
* Mixed bed	300	50% CO_2/50% Air	99·24	2·00	45 g Ag/Cu 153 g Carbon	62·0
BCNU.CP only	300	50% CO_2/50% Air	99·51	2·00	186 g Carbon	—
S & S only	300	50% CO_2/50% Air	99·89	2·00	170 g Carbon	—

*=BCNU grade CP charcoal and silvered copper mesh (ex. Hydronyl Syndicate).

Closed-circuit Television at Trawsfynydd

by I. M. WATERS and P. BARRATT, M.A., Ph. D
(*Pye TVT Ltd.*)

THE potential utility of closed-circuit television techniques in nuclear work scarcely needs emphasizing. Its use at Trawsfynydd station, under construction by the A.P.C. consortium, is an excellent example of the application of standard equipment as an integral feature of the design—it is only too easy to attempt to add television equipment as an afterthought, and find that requirements are either exacting or impossible.

A basic camera chain, as shown in Fig. 1, consists of four units; the camera itself, the CCU (camera control unit) the CCP (camera control panel) and the picture monitor.

Basic Units

The camera is kept as small as possible for two reasons; first because space is often at a premium and, second, because in unfavourable environments it is advisable to keep the amount of circuitry down to a minimum. It consists therefore of an optical system, video tube with deflection and focus components, and a 2-valve head amplifier; all controls and supplies are carried by a single cable, which may be up to 1 000 ft long.

The preferred operating system for industrial use is 625 lines, 50 fields, with interlaced scanning. The spectral response of the camera tube is approximately the same as panchromatic film, and light levels should, ideally, be 50 foot-candles (500 lux), which is not abnormally bright for industrial work. Satisfactory results can be obtained with less than 10 foot-candles if an f/1·9 lens is used.

Optical focusing of the camera may be either manual or remote; it is also possible to fit remotely controlled lens change or a zoom lens. Remote operation of pan and tilt heads is also possible.

The camera control unit, which may be either portable or rack-mounted, does not necessarily have to be mounted near the operator, since it does not carry any actual controls. It contains a video amplifier which corrects the frequency characteristics of the camera signal, and adds the necessary blanking and synchronizing wave-forms. It is also provided with an automatic sensitivity circuit, to compensate for changes in object lighting, if required.

Where it is necessary to install a number of cameras, and simultaneous viewing is not required, it is possible to fit a switching unit to enable one CCU to control up to five cameras.

The CCP contains the controls for the chain, including auto/manual sensitivity controls, beam and focus controls and switches for optical focus and lens change.

Trawsfynydd Scheme

The Trawsfynydd scheme comprises, in effect, two separate installations; one for the charge machines, charge chute test rig and irradiated fuel disposal room, and another separate

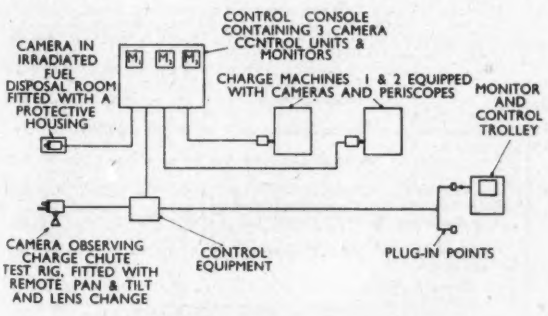
CONTROL CONSOLE CONTAINING 3 CAMERA CONTROL UNITS & MONITORS
M1 M2 M3
CAMERA IN IRRADIATED FUEL DISPOSAL ROOM FITTED WITH A PROTECTIVE HOUSING
CHARGE MACHINES 1 & 2 EQUIPPED WITH CAMERAS AND PERISCOPES
MONITOR AND CONTROL TROLLEY
CAMERA OBSERVING CHARGE CHUTE TEST RIG, FITTED WITH REMOTE PAN & TILT AND LENS CHANGE
CONTROL EQUIPMENT
PLUG-IN POINTS

Fig. 2.—Block diagram of system (excluding control rod service bay).

Fig. 1.—General view of basic unit (camera, monitor, CCP and CCU).

installation for the control rod servicing bay. A block diagram of the former is shown in Fig. 2.

Each of the charge/discharge machines is provided with a camera with remote-focusing facilities. These observe fuel elements and the operation of internal mechanisms via periscope systems; there are, therefore no radiation problems, and ambient conditions are normal.

A single camera is employed to obtain close-up views of the fuel chute under test and for general observation over the working space. The camera is fitted with remote focus and lens-change, and is mounted on a light-duty pan and tilt head carried in a movable structure to ensure full azimuthal coverage.

There may be low level radiation fields arising from fuel chutes which have been inserted in the reactor but these will have no significant effects upon the equipment.

The camera is mounted in a housing on a "bottling machine" and is intended to view the insertion of fuel elements into bottles and the subsequent lidding of the bottles. A fixed-focus camera is adequate and a sealed housing is provided by Fairey Engineering, Ltd., to overcome the possible effects of humidity.

The expected gamma radiation levels lie between 10^6 and 10^8 r/year. This is sufficiently high for radiation damage effects to be taken into consideration, the most susceptible components being the camera tube and the optical system, both of which show appreciable damage when doses of some 10^8r have been received. It is likely that replacement of these items will be necessary at intervals of approximately one year. In the case of the camera tube, this time will probably be longer than the life to be expected under normal conditions (depending, of course, upon the frequency of use) and radiation damage should not be the limiting factor as regards life or replacement interval.

Control Rod Service Bay

At this station provision is made to examine closely control rod assemblies taken out of the reactor, by lowering them past groups of cameras located within the servicing bays. This is the most complex part of the installation, as observations must be made at three different levels within each of four bays.

The proposed layout of the cameras in any one bay is shown in Fig. 3. At the lowest level, cameras are provided to give a magnified view of the entire circumference of the control rod as it is lowered from the guide tube, so that cracks and surface defects may be detected at an early stage.

As full circumferential viewing is required and the camera

installation is confined to an "L" shaped space it has been found necessary to use a combination of mirrors and cameras. The control rod is at the corner and two cameras are in either arm. Two vertical mirrors, at right angles to each other, are positioned to enable two of the cameras to observe the backs of the control rods; while the other two cameras observe the front.

The camera outputs are scanned sequentially by an electronic switch so that a single monitor displays a succession of views of the control rod, each lasting from 0·5-2 seconds, the control rod descending a few inches per second during viewing. A "stop" button stops both control rod winch and electronic switch action, if the operator sees any feature requiring closer study, and manual "inching" is then possible, together with changes in lighting aspect to bring out details more clearly.

The cameras are manually focused and are operating in a normal environment. The anticipated gamma dose rate of 10^4-10^5 r/year should not be significant.

At the intermediate level, each of the four servicing facilities has a camera permanently fixed within the bay for viewing the operation of replacing a control rod cable.

At the upper level, however, the cameras are fixed to a transportable mounting provided by Fairey Engineering Limited, so that they may be moved to any of the four servicing facilities. Cables are plugged into termination boxes installed in each bay. These four cameras are located at the corners of a square, with the object at the centre, and are scanned sequentially as at the lower level. Upper level viewing is required to inspect the control rod guide tube. Again, viewing is required over the entire circumference and a detailed examination is necessary. Since two different diameters are to be inspected a lens change unit is provided so that the image always fills the monitor screen, giving maximum resolution. Expected radiation levels are 10^3-10^4 r/year and can be disregarded.

Control and Display Facilities

The charge machine, irradiated fuel disposal room, and the charge chute test rig cameras are all connected to a control console carrying three monitors, camera control panels, and pan and tilt controls for the charge chute test rig installation. Two of the monitors are associated with the two charge machine installations. The third monitor displays either the irradiated fuel disposal room or the charge chute test rig; selection is by means of a video selector switch on the console.

Charge chute test rig signals are also brought out to two separate cable termination boxes. A trolley containing all the necessary manual controls for the camera can be connected at either point. The use of either trolley position, or the passing of signals to the control console is determined by the setting of a 3-way switch in the charge chute test rig control room.

The trolley carries a 14-inch monitor and can be moved about the charge face on rubber-tyred wheels. It serves the additional function of being a mobile, spare display and control facility if any other part of the system should fail.

The control and servicing facilities for each reactor involve four control-rod service bays; each contains a group of cameras, with viewing at three levels. The four groups are arranged in two sets of two. One monitor is provided for each set and the display of the presentation is governed by a rotary electronic switch and a 3-way composite camera switch, which enable the upper or lower levels to be selected.

Since only one of the two sets of bays is in use at any time arrangements are made for control and observation to be carried out from the adjacent bays, in order to remove the operators from the radiation hazard area.

In addition to the sequentially scanned cameras the other cameras provided at the lower levels to observe the hoist cables and the attachment of thermocouples may be selected by the camera switch.

Acknowledgments

Thanks are due to the Directors of Pye TVT Ltd. for permission to publish this article and to Atomic Power Constructions Limited and Fairey Engineering Limited for allowing us to include design information necessary to the description of the television systems, and for help received in the final stages.

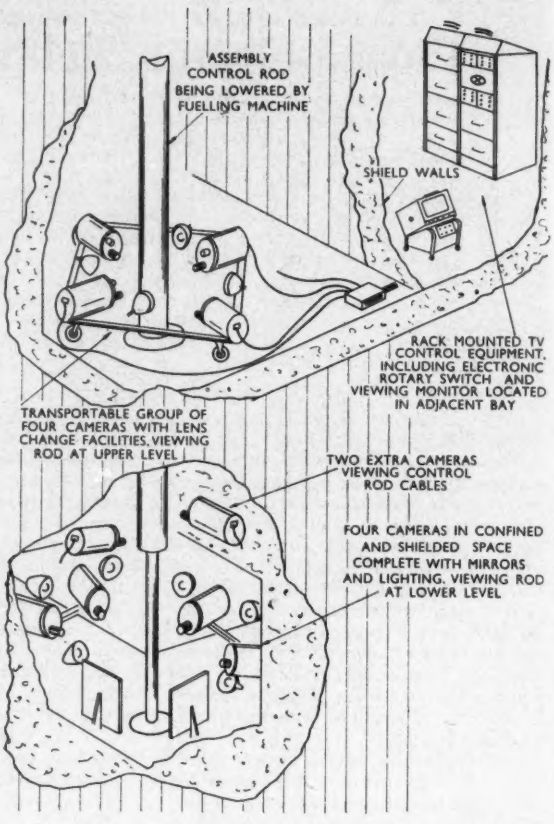
ASSEMBLY CONTROL ROD BEING LOWERED BY FUELLING MACHINE
SHIELD WALLS
RACK MOUNTED TV CONTROL EQUIPMENT, INCLUDING ELECTRONIC ROTARY SWITCH AND VIEWING MONITOR LOCATED IN ADJACENT BAY
TRANSPORTABLE GROUP OF FOUR CAMERAS WITH LENS CHANGE FACILITIES, VIEWING ROD AT UPPER LEVEL
TWO EXTRA CAMERAS VIEWING CONTROL ROD CABLES
FOUR CAMERAS IN CONFINED AND SHIELDED SPACE COMPLETE WITH MIRRORS AND LIGHTING. VIEWING ROD AT LOWER LEVEL

Fig. 3.—Arrangement of cameras in control rod service bay.

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NEW BOOKS

Nuclear Reactor Instrumentation. By M. W. Jervis, M.Sc. Tech., A.M.I.E.E. (Senior Engineer, Electrical Department, The Nuclear Power Group). (74 pp., Temple Press Limited, 12s. 6d.)

Nuclear instrumentation, if not a separate science, is a specialized branch of nuclear engineering and reactor instrumentation is even more highly specialized. It is difficult, in fact, for the novice, or for those with only the most superficial knowledge of the subject, to know where to begin.

Fortunately, the *Nuclear Engineering* Monographs Series was evolved for just such situations—to aid those whose main interests lie outside a particular subject—and this example is no exception. Since it would be ridiculous to suggest that reactor instrumentation could be dealt with at any length in the compass of a 74-page book, the author has, very wisely, confined himself to the broad outlines of his subject, with copious references to other published work for those who wish to pursue a particular item in greater detail.

A brief introduction to reactor kinetics explains the importance of instrumentation and the functions which lend themselves to measurement; temperature measurement and flux measurements are then reviewed, followed by the general principles of safety systems. The measurement of flux distribution within the core (i.e., flux plotting) is followed by a chapter devoted to burst cartridge detection. The principles of telemetering flow measurement and gas analysis are reviewed, and the final chapter is devoted to health physics instrumentation. As previously stated, there is a very full list of further reading for each chapter, more than 100 references being listed. There is also an adequate index.

Alloy Steels. Published by Samuel Fox and Co., Ltd., Stocksbridge Works, Sheffield. 42s.

At first glance, this publication with its loose leaf format, and limp leather binding, appears to be a company publication; closer examination will show that it is not. It is, instead, a collection of technical data on a selection of constructional alloy steels set out in a variety of ways so as to be of maximum use to metallurgists. In addition to data showing the effects of alloy composition, of mass and of heat treatment on the mechanical properties of steel, it includes information on the newer concepts of hardenability, isothermal and continuous cooling transformation; many of the transformation diagrams are in colour. The whole publication represents one of the most comprehensive collections of technical data available on British Standard alloy steels.

Thermoelectricity: Science and Engineering. By Robert R. Heikes and Roland W. Ure, Jr., and 14 collaborators. (476 pp., Interscience Publishers, Ltd., London. 139s.)

With the growing interest in thermoelectric generation, this book is timely. As is the case with most books dealing with a complex subject nowadays, specialized sections are written by different individuals. Contents include classical and irreversible thermodynamic treatment of thermoelectricity; semiconductors and semi-metals; narrow band semiconductors, ionic crystals, and liquids: thermal conduction in thermoelectric materials; radiation effects; diffusion; crystal growth; the chemistry of mixed valence materials; measurements of properties of thermoelectric materials; theory of thermoelectric materials; low temperature Peltier cooling; survey of known thermoelectric materials; thermionic conversion; theoretical calculation of device performance; some heat transfer problems in thermoelectric devices, and technology of thermoelectric devices.

Encyclopædic Dictionary of Physics: Vol. 1. Editor-in-Chief J. Thewlis, Harwell; Associate Editors: R. C. Glass, D. J. Hughes, and A. R. Meetham. (800 pp., Pergamon Press, London. £80 per set.)

This is the first volume of a work with the avowed object of putting the whole of physical knowledge on to a bookshelf; it is nearly 40 years since the Dictionary of Applied Physics appeared in English. It is being produced with an international board of more than 60 consulting editors, and has been carefully compiled, so that while alphabetical in arrangement, there is the minimum of cross-referencing. Although comprehensive, references and bibliographies are given for further reading if required. The first volume, covering Abbé Refractometer to Compensated Bars runs to 800 pages, and is the work of some 440 contributors.

Reactor Handbook, Vol. 2. Fuel Reprocessing. Edited by S. M. Stoller and R. B. Richards. (665 pp., Interscience Publishers, Ltd., London. 158s.)

This publication is volume 2 of the second edition of the Reactor Handbook, which was first published at the time of the 1955 Geneva Conference, fuel reprocessing being the only operation which was not declassified at the time. It has now been revised and brought up to date and is divided into six sections: introduction; aqueous separations, non-aqueous separations (pyrometallurgical); reconversions; waste disposal; engineering.

Steels for Reactor Pressure Circuits. (587 pp., The Iron and Steel Institute. 65s.)

This book represents the collated papers from a symposium held in December, 1960, by the Iron and Steel Institute for the B.N.E.C. The six sessions covered by the book are high-temperature properties; corrosion; fabrication aspects; irradiation effects; steels for future reactors; general discussion.

Electronics: A Bibliographical Guide. By C. K. Moore and K. J. Spencer. (411 pp. Macdonald and Co. (Publishers), Ltd., London. 65s.)

It is scarcely necessary to emphasize the literally fantastic speed at which information on electronics appears; for example, it is estimated that 1,500 papers on semiconductors alone appear each year. Here, the bold step has been taken of producing what might be termed a "second differential" of the field. In each of the sections, the appropriate specialist bibliography is quoted; in other words, the work is in part a bibliography of bibliographies. In addition, it lists authoritative text books, survey articles, main conventions and conferences, and a wealth of papers in which have been presented significant advances in the science, together with numerous tutorial papers. Throughout, the aim has been to provide a means by which any paper in the now vast literature of electronics may be located.

B.S. 3385 : 1961. Specification for Direct-Reading Personal Dosemeters. (British Standards Institution, 3s.)

This is the first British Standard to be prepared under the authority of the newly formed Nuclear Energy Industry Standards Committee of the B.S.I. It lays down standards for the design, range, scale, etc., of the instruments and limits of reproducibility, etc. Attention is drawn to the effect of beta radiation and recommendations are made for the appropriate sleeves to compensate for this.

Other new British Standards recently issued include **B.S. 2972 : 1961,** Methods of test for Thermal Insulating Materials, and **B.S. 131 : Part 1 : 1961,** Notched bar tests: Part 1; The Izod impact test on metals.

Books Received

Fuel Element Fabrication, Volume 2. (Proceedings of a Symposium in Vienna, May 10-13, 1960.) Edited by the International Atomic Energy Agency. (384 pp., Academic Press Inc. (London), Ltd. 71s. 6d.)

The Plasma State. By E. J. Hellund. (197 pp., Reinhold Publishing Corporation, New York. 52s.)

Protection Against Radiation. By John D. Abbatt, J. R. A. Lakey and D. J. Mathias. (235 pp., Cassell and Co., Ltd., London. 25s.)

Proceedings of an International Conference on **Instrumentation for High-energy Physics,** Berkeley, U.S.A., 1960. Edited by C. E. Mauk, A. H. Rosenfeld and R. K. Wakerling. (321 pp., Interscience Publishers, Ltd., London. 75s.)

Radiation Chemistry of Gases. By S. C. Lind. (313 pp., Reinhold Publishing Corporation, New York. 100s.)

MEETINGS

October 17.—Society of Instrument Technology (Bristol Section), in the Bristol University Department of Physics, at 7.30 p.m. **Instrumentation and Control at the Hinkley Point Nuclear Power Station,** R. B. Quarmby.

October 19-20.—Institution of Chemical Engineers (Graduates and Students Section), at the Royal Overseas League, London, S.W.1. A symposium: **Scale-up and Pilot Plants.**

October 26.—Society of Instrument Technology (Chester Section), at Stanley Place, Watergate Street, Chester, at 7 p.m. **Radiation Pyrometry,** T. Land.

October 31.—Institution of Chemical Engineers (South Wales and Monmouthshire Group), at South Wales Institute of Engineers, Cardiff, at 7.15 p.m. **Ion Exchange in the Atomic Industry,** T. V. Arden.

November 13.—The Institute of Physics and Physical Society with Society of Instrument Technology, at Manchester University. **The Role of the Physicist in Plant Instrumentation,** L. J. R. Postle.

November 16.—Society of Instrument Technology (East Midland Section), at The College of Further Education, Loughborough, at 6.45 p.m. **A Survey of the International Temperature Scale,** J. A. Hall.

November 28.—B.N.E.C., at the Institute of Marine Engineers at 5.30 p.m. (sponsored by the Joint Panel on Nuclear Marine Propulsion). **High-temperature gas-cooled reactors for marine propulsion,** R. E. Richards.

November 30-December 1.—B.N.E.C., at the Institution of Electrical Engineers. Symposium: **Nuclear Electronics.**

Industrial Notes

Members of the Japanese Atomic Industry Structure Study Team inspect a model of the Tokai Mura reactor core set up at the GEC Erith works for studying earthquake effects. The team, sponsored by the Japanese Atomic Industrial Forum and the U.S.-backed International Cooperation Administration, were visiting nuclear industrial organizations in the U.K. and Continent following a tour of the United States.

The withdrawal of **Crompton Parkinson** from **APC** has been made complete by the announcement of the resignation of Mr. J. V. Daniel, Crompton Parkinson managing director, from the consortium board. A founder member of APC and supplier of switchgear and transformer equipment for Trawsfynydd, Crompton Parkinson feel that the business is not adequate to justify their remaining as partners and particularly in a group as large as United Power Company. This feeling was, in fact, made known before negotiations between APC and GEC, parent companies of UPC, started and had nothing to do with the temporary suspension of these negotiations. All the company's commitments for Trawsfynydd will, of course, be fulfilled.

J. and S. Pumps are to supply glandless circulating pumps for the ENEA Dragon project. Value of the contract is about £19 000.

Dewrance and Company are to supply all boiler mountings and steam and feed pipeline valves for the Sizewell nuclear power station being built for the CEGB by the English Electric, Babcock and Wilcox, Taylor Woodrow Atomic Power Construction Company.

A symposium on photomultiplier tube applications was held by **EMI Electronics** at their London offices, September 13-15. Papers on subjects ranging from nuclear physics to seed sorting, from astronomy to printing were presented. The meeting, which ended with a visit to the company's works at Hayes, Middlesex, was well supported.

Mervyn Instruments, of Woking, Surrey, have followed **West Instrument,** of Brighton, in becoming a member of **Gulton Industries (Great Britain),** the wholly owned subsidiary of Gulton Industries Inc. of America. Mervyn Instruments specialize in analytical chemical process instruments and have supplied various equipment and apparatus for nuclear installations both in the U.K. and overseas. West Instrument, who were the first to join Gulton Industries, specialize in temperature control.

Motors and control gear for boiler feed and c.w. pumps in the Dungeness nuclear power station are to be supplied by the motor and control division of **AEI,** who will also provide control centres for fans and valve motors used elsewhere in the plant. Total value of the orders for this equipment is £271 000.

Health physics equipment used in conjunction with the loading of No. 1 reactor at Berkeley includes £40 000 worth of monitors supplied by **EMI Electronics.** These are of both the hand and clothing and portable type. Similar equipment is being supplied for Hinkley Point.

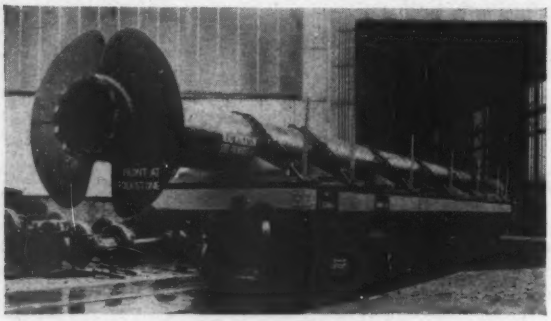
A **European Computer Manufacturers' Association** with headquarters at Geneva has been formed. Its purpose is to establish data processing standards. Three working committees have already been established and a fourth is in the making. All principal U.K. computer manufacturers have joined.

The British Aluminium Company have secured full control of **Magnesium Elektron,** having bought the 40% interest previously held by **The Distillers Company.** Magnesium Elektron in which BAC had already had a 60% interest, is the sole producer of magnesium metal in the U.K. and a leader in the development of magnesium alloys.

The fifth advanced course in "The Principles of Radiation Protection" to be organized at the **Harwell Reactor School** starts on October 25. It closes on January 30, 1962.

The Plessey Company have secured the lease on a 50 000 ft² factory at Cheney Manor, Swindon, close to their existing plant. It is expected to be used for the expansion of the company's industrial hydraulics division.

A former Masonic Temple in Broad Street, Birmingham, is being converted into an exhibition hall to house the **Engineering and Building Centre,** which is to be moved from Stephenson Place. Conversion is costing £80 000 and is expected to be finished some time early next year. Covering 37 000 ft², the new centre will have space for about 600 displays.



One of the 12 cast steel tube poles supplied by Head Wrightson Steel Foundries for the Dungeness nuclear power station c.w. intake structure.

Eutectic Welding Alloys celebrated in August the 10th anniversary of the formation of their British company. They claim to be the only company capable of producing electrodes right through all stages from the raw metals to finished product, in *any* metal.

George Kent have received an order worth more than £32,000 for instrumentation at Trawsfynydd.

Atlas Copco have established an international advice bureau for dealing with compressed air problems in civil and mining engineering. The address is 11 Clarges Street, London, W.1, and the telephone number Grosvenor 1378.

The commercial divisions and most of the production staffs of **Mullard Equipment** have moved to their new factory at Manor Royal, Crawley, Sussex (telephone No.: Crawley 26386).

Semiconductors, of Swindon, is now a wholly owned subsidiary of The Plessey Company, the Philco Corporation, U.S., having sold its minority share in the company.

A course of six lectures on "Compact Heat Exchangers" starts at the **National College for Heating, Ventilating, Refrigeration and Fan Engineering,** London, on October 23. Fee for the course, which is intended for graduates, is 10s. Details are obtainable from the clerk to the governors.

Vactric Control Equipment, which was recently taken over by the News of the World Organization, have acquired a site in Morden, Surrey, for building a development and applied research department.

Delivery of the two secondary steam generators for the 150 MW Garagliano plant, Italy, is expected to be made mid-October. The units have been fabricated in the Netherlands by **Koninklijke Machinefabriek Gebr. Stork and Co.,** Hengelo, a member of **Neratoom.**

Tracerlab have received a $90 000 contract for designing and manufacturing monitoring equipment for the NPR.

DSIR have moved their headquarters to State House, High Holborn, London, W.C.1. Telephone number is Chancery 1262.

The Atomic Industrial Forum, U.S.A., has moved its headquarters to 850 Third Avenue, New York, 22, N.Y.

Personal

Mr. P. T. Fletcher

Mr. L. J. Davies

Dr. J. E. Stanworth

Capt. G. R. B. Pattison

Appointments—U.K.

Mr. P. T. Fletcher, special adviser to the AEA Member for Reactors and, until the reorganization of the Authority, deputy managing director of the development and engineering group, Risley, as a director of construction and supply for the United Power Company.

Mr. L. J. Davies, director of research, AEI (Rugby), as director of research for the group, with overall responsibility for the AEI research establishments at Aldermaston, Harlow, Rugby and Manchester. He also becomes a non-executive director of AEI (Rugby). New director of research for this company is **Dr. J. E. Stanworth,** who has been in charge of research on new materials, particularly for use in semiconductors and nuclear power.

Captain G. R. B. Pattison, assistant director of electrical engineering with the Director-General Ships, Admiralty, Bath, as deputy general manager of Plessey Nucleonics.

Mr. H. G. Smith, of the chemistry division, Oak Ridge, to the AERE, Harwell, for one year. He is working with the neutron diffraction group in the solid state physics division.

Sir William Holford, professor of town planning, University College, London, as part-time member of the CEGB for another term of office.

Mr. S. J. Reason, formerly senior technical representative with Nordac, chemical engineers, of Uxbridge, as manager of Uddeholm's newly created chemical and nuclear plant division (see "Industrial Notes," September, 1961).

Dr. T. P. Hughes, director of research, Tube Investments, as joint assistant managing director of the B.B. Chemical Company. New TI director of research, as announced earlier this year, is **Dr. J. M. Kay,** formerly professor of nuclear engineering at Imperial College of Science and Technology, London.

Mr. E. J. Robinson, Middle East manager for the Head Wrightson Export Company, to the board, and **Mr. R. F. N. Otway,** from the Morgan Crucible Company, as European manager.

Sir John Cockcroft, as president of the British Association for the Advancement of Science, 1961-62.

Mr. Robert Butler, a director of Quasi Arc, as managing director of the Eutectic Welding Alloys Company.

Mr. E. J. Aaronson, overseas economic and financial adviser to GEC, England, as general manager, overseas operations.

Mr. Cecil F. Hurst, deputy chairman of Samuel Osborn and Company, as assistant managing director.

Mr. R. H. Dent, deputy chairman and managing director of the Cape Asbestos Company, as chairman of Cape Insulation and Asbestos Products, and **Mr. A. G. Grant,** managing director of CIAP, as chairman director of CIAP, as chairman of Andersons Insulation Company.

Mr. F. L. Tombs, sales manager of the GEC power plant division, Erith, as manager, engineering services, with responsibilities for sales, tendering, contracts and operational services departments for site resident engineers. **Dr. D. Kalderon,** chief development engineer, as deputy chief engineer of the division; **Mr. W. G. Rhodes,** as home sales manager, and **Mr. S. D. Thorpe,** as overseas sales manager.

Mr. W. P. Raffan, responsible for the development of solid state devices at Rank Cintel, as head of the newly formed solid state division of 20th Century Electronics.

Overseas

Dr. R. A. Charpie, an assistant director of Oak Ridge and director of the EGCR programme, as manager of advanced developments for the Union Carbide Corporation. His successor on the EGCR programme is **Mr. W. D. Manly,** previously working on the aircraft nuclear propulsion project and GCR materials studies.

Mr. John A. Erlewine, formerly special assistant to the assistant general manager, research and industrial development, U.S.AEC, as senior AEC representative in the U.S. Mission to the European Communities, Brussels, and **Mr. Allan M. Labowitz,** assistant to the director of reactor development, AEC, as scientific representative. Mr. Erlewine succeeds **Mr. Nelson Sievering,** who had been senior representative since 1959 and is now associate director for advanced systems division of reactor development, AEC. In Brussels, Mr. Erlewine will act as deputy for Euratom affairs to the head of the Mission, **Mr. Walton Butterworth.**

Mr. Leonard J. Linde, Allis-Chalmers director of engineering services, as director of engineering of the company's industries group, and **Mr. Dane T. Seag,** technical director of C Stellerator Associates, as assistant director of research in the research division. C Stellerator Associates was a group formed by Allis-Chalmers and RCA for building the C Stellerator at Princeton University. From 1957 until 1960 it was managed by Mr. Linde.

Mr. Mansfield D. Sprague, vice-president and director of AMF, as president and chief executive of AMF Overseas, Geneva, and **Mr. Frederic Chapuisat,** a director of the parent company, as general manager.

Mr. Carl F. Leyse, of General Atomic, as general manager of the Internuclear Company. He was with the company before going to General Atomic, from 1955-59, and before that was chief of the reactor projects engineering section, Phillips Petroleum Company.

Dr. Robert L. Loftness, Atomic International's senior representative attached to Dyanatom, Paris, returns to the States as AI's Eastern representative with headquarters in Washington. He succeeds **Mr. Robert H. Sehnert,** who has been assigned to advanced planning at Canoga Park, California.

Mr. N. Pelser, head of the reactor department, Reactor Centrum Nederland, as technical managing director of the organization. He succeeds **Mr. J. J. van Rijsinge,** who was appointed in 1959.

Resignation

Mr. J. V. Daniel, managing director of Crompton Parkinson, has resigned from the board of Atomic Power Constructions. His resignation follows the withdrawal of his company from the consortium.

Mr. H. G. Smith

Mr. S. J. Reason

Dr. R. A. Charpie

Mr. W. D. Manly

Technical Papers and Publications

The Institution of Mechanical Engineers

Lubricants for Nuclear Reactors. By P. E. Vaile, A.M.I.Mech.E. (Mobil Oil Co., Ltd.). (Paper sponsored by the Lubrication Group, in conjunction with the Nuclear Energy Group.)

It was considered, at one time, that lubrication would be impracticable under conditions of high temperatures and irradiation levels, and the earlier gas-cooled reactors were designed to operate without lubrication. Since then, a great deal of work has been done, and experience obtained, regarding the properties of lubricants under simulated operating conditions, not only from the standpoint of deterioration of the lubricant itself, but also having regard to the compatibility problems with Magnox canning.

The paper deals chiefly with gas-cooled graphite-moderated reactors, although other types are briefly surveyed; CO_2 is assumed to be the normal coolant although consideration is also given to helium. In addition to the problems of actual lubrication, the problems attached to the use of hydraulic fluids are studied.

In considering gas-cooled reactors, the paper lists the operating conditions for the various parts of the reactor and its associated gear (i.e., blowers, gas valves, control rod mechanisms, fuelling machines, auxiliary equipment, etc.) with regard to radiation dosage, operating temperature and compatibility problems, if any. Blowers, for example, are not exposed to direct radiation and, therefore, from the radiation standpoint, a high-quality turbine oil would operate satisfactorily, since the dose from coolant activity and circulated fission fragments would probably not exceed 10^3 rad/year. Since, however, minute quantities of oil are bound to enter the CO_2 circuit, compatibility must be watched. Oxidation stability is another important point for blower lubricants; the seal will operate at higher temperatures than the bearings, particularly if allowance is made for "soak-back" during shut-down. Gas temperatures, too, are on the increase. The U.K.AEA requirements for various types of oil and grease are given in detail.

Methods of testing lubricants are reviewed. It is pointed out that gamma irradiation is preferred for test purposes, for two reasons; first that it has been shown to be the predominant form of radiation in the areas where lubrication is required and, secondly, because of its convenience in testing. Unlike pile-irradiated materials, gamma-irradiated materials can be handled immediately after irradiation and larger quantities can be handled at a time. The use of the irradiation pond at Harwell is described, together with a number of different test rigs (one of these, the SETRA grease-testing rig, was described in *Nuclear Engineering* for December, 1960, p. 591).

As might be expected, the effect of radiation on lubricants, from the standpoint of actual physical and chemical changes, is given detailed consideration. Cross-linking, unsaturation and gas production are illustrated, and the conception of "anti-rads" or the use of exceptionally stable substances (such as synthetic aromatics) as additives, is explained.

Grease failure takes two forms; destruction of the thickener (softening of the grease) and degradation of the fluid (solidification). The precise pattern of consistency change is, therefore, dependent on the relative radiation stability of the two. This can be misleading with tests at a single irradiation dose instead of a range; an example is given of a grease which first softened due to thickener breakdown (at about 100 Mrad) and then hardened due to polymerization of the fluid and, at 170 Mrad, showed its original consistency. Had a single test been taken at 170 Mrad, it would have appeared unaffected.

A long section of the paper is devoted to the characteristics of special nuclear lubricants, and the precautions to be observed in selecting conventional lubricants, and a further section covers the use of oil-impregnated sintered iron bearings, molybdenum disulphide and dry running. It was concluded that, with the trends towards advanced designs with more severe operating conditions, it might still be necessary to resort to dry running for extreme conditions.

U.S.AEC

BNL 661 (T-215). Table of Reactor Coolant Properties. By Leon Green, Brookhaven National Laboratory. (Office of Technical Services, U.S. Department of Commerce, Washington 25, D.C. $0.50.)

A very useful quick reference to the important properties of what might be called "practical" coolant materials. In addition to the physical, thermal and nuclear properties, additional general information, such as approximate cost, container requirements, fire and explosion hazards and reaction with uranium and plutonium, is given for each type of material.

Under liquids are listed water, heavy water, diphenyl, Dowtherm-A and Santowax R. The table of gases includes hydrogen, helium, CO_2, air, nitrogen, steam and neon. Liquid metals is a formidable table, including bismuth, gallium, lead, lithium, mercury, potassium, rubidium, sodium, tin, zinc, two varieties of NaK (including the eutectic) and the lead-bismuth eutectic. The final table covers properties of fused inorganic material, including some of the complex fluorides. The report is complete with references and bibliography.

TID-7603. Proceedings of the Uranium Carbide Meeting held at Oak Ridge National Laboratory, Dec 1-2 1960. (Office of Technical Services, U.S. Department of Commerce, Washington 25, D.C. $1.75).

This report, which runs to more than 160 pp, contains the following papers:—

Uranium-plutonium carbide fabrication at Argonne.

Progress in uranium carbide technology at Atomics International.

Research on uranium carbide and uranium carbide-base fuel materials at Batelle Memorial Institute.

Research on uranium carbide at Brookhaven National Laboratory.

Status of carbide fuel development at the Carborundum Co.

Carbide research at General Atomic.

Carbide research at Lawrence Radiation Laboratory.

Studies within the system UC-UC_2 at Los Alamos Scientific Laboratory.

Development of uranium carbide production processes at Mallinckrodt Nuclear.

NASA research on uranium carbide and refractory ceramics.

Graphite-matrix nuclear fuel element development at the National Carbon Co.

Carbide fuel development at Nuclear Development Corporation of America.

Uranium carbide research at ORNL.

Development of uranium carbide as a nuclear fuel at Olin Mathieson Chemical Corporation.

Progress in technology of columbium alloy-clad uranium carbide fuel elements at Union Carbide Metals Co.

The preparation of uranium monocarbide by reaction sintering at AERE, Harwell.

Advances in uranium carbide technology in France.

Experimental work on uranium carbide at Euratom-Nukem.

U.K.AEA

List of Publications Available to the Public. Cumulation No. 5 (January-December, 1960). Library, Research Group, U.K. AEA, Harwell. (H.M.S.O., 17s. 6d.)

As its title implies this is a list of all unclassified reports published during the period, either directly or in the form of published articles. The arrangement is very useful for rapid reference, the grouping being as follows:—

General; biology, medicine, agriculture; chemistry, general, radio, radiation, nuclear, analytical; chemical engineering; engineering; instrumentation; geology and mineralogy; health and safety; industrial applications of isotopes and radiation; isotope separation; mathematics and computers; metallurgy and materials; radiation effects; physics, general, solid-state, nuclear and theoretical; plasma physics and thermonuclear devices; particle accelerators; reactor physics; reactor technology; waste disposal.

Patents and reports which have appeared only in the published literature are separately grouped; there are also author and serial indexes.

AERE-M 885. A technique for brazing stainless-steel holders containing cobalt-60 radioactive sources. E. Brown and J. G. Purchas (Metallurgy Division) AERE, Harwell. (H.M.S.O., 2s. 6d.)

Source holders are in two parts—a thimble to hold the active material and a stainless-steel shank, the two being screwed together and brazed. Induction brazing is used, with a wire of SCP-6 alloy (Cu 82% Pd 18%) supplied by Engelhard Industries, Ltd. For low-activity sources (below 3 mc) a number are brazed together on a turntable arrangement; sources greater than 10 mc are handled separately. This report details the arrangements for brazing *in vacuo*, and the shielding and viewing arrangements adopted; it also gives an account of the optimum conditions (number of turns in coil, current, time, etc.), to provide brazing of first-class quality.

Processes and Equipment

For further information on any item please enter the relevant number on one of the Reader Service Cards enclosed with this issue and forward to the address given.

High-temperature Flowmeter

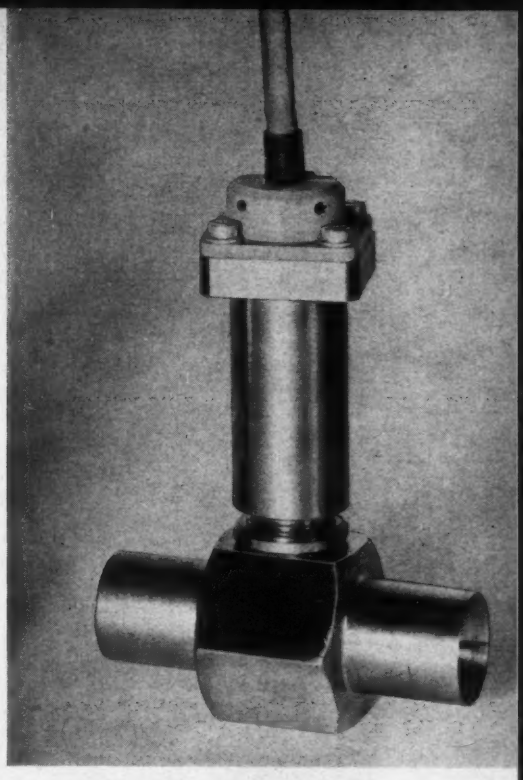
Accurate metering of the flow of high-temperature water or organic liquids such as terphenyl has frequently presented some difficulty. A new Meterflow unit, developed especially for the nuclear field, is capable of operation up to 400°C (753°F).

The meter operates on electromagnetic principles, somewhat on the lines of an inductor alternator. A rotating impeller, driven by the fluid, carries an inductor of slightly magnetic stainless steel; the main flux is supplied by a stationary permanent magnet which, with the inductor coil, is located externally to the main flow pipe; the frequency of the induced e.m.f. is a measure of the impeller speed. A noteworthy feature of the design is the method of compensation for the increased friction which is caused, at high flow rates in this type of instrument, by the pressure of the impeller against the downstream bearing. This takes the form of a bled flow through the impeller shaft (proportional to the pressure drop across the impeller) to act against a small boss on the shaft, thus relieving the end pressure, and allowing an overall linearity of better than 1% with a "repeatability" of better than 0·2%. The unit is suitable for working pressures up to 3 000 psi, and is available in three ranges. 0·3–3, 1–8 and 3–25 gal/min. All three ranges have working pressure drops of about 5 psi. The output frequency at full load is 1 000 c/s.

The body of the meter is of stainless steel, and is suitable for welding into rigs. Bearings are of various materials according to the operating temperature; for 400°C service a silver-graphite compound is used.

(Meterflow, Ltd., North Feltham Trading Estate, Feltham, Middlesex.)

1138



The high-temperature Meterflow unit.

Vacuum Furnace Loader

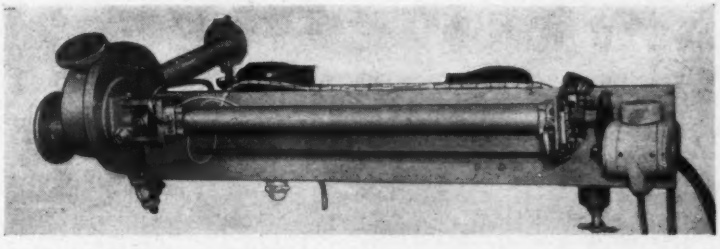
Sintering of fuel pellets in a vacuum furnace is assisted by an automatic loading mechanism, developed for the AEA, which allows individual loading of up to 36 pellets, under timed sequence control, without breaking vacuum.

A removable magazine, after loading in a glove box, is placed into the loading unit, which is mounted on the furnace. The door at the inner end of the magazine is automatically opened, and the furnace raised to its working vacuum and temperature. The mechanism contains a long arm, with a cup at the end which, after picking up a pellet, is pneumatically extended into the furnace through a water-cooled seal. At the end of its travel, it rotates through a half-turn to deposit the pellet. A push-plate is fitted at the end of the arm to prevent pellets piling up in the furnace.

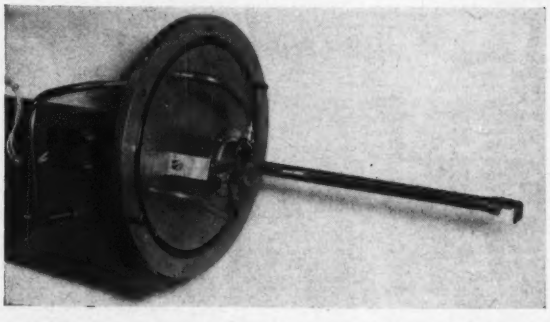
The entire sequence, from the loading of the magazine, is under the control of a programme unit, which counts the pellets and gives visible indication of the count, and of the completion of the loading. Failure (e.g., loss of operating air supply leaving the long arm in the hot zone) not only sounds an alarm, but withdraws the long arm by means of an auxiliary air reservoir.

(Spembly, Ltd., New Road Avenue, Chatham, Kent.)

1139



Above: General view of the Spembly furnace loader.

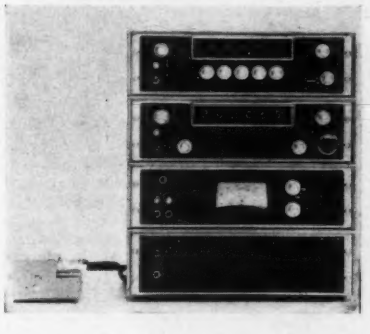


Left: End of unit, with magazine unit removed and long arm extended.

New U.S. Instruments

Nuclear-Chicago announce the first four of a new range of nuclear instruments in chassis form suitable for use either singly or arranged in an integrated unit called the Logic 801. The four units are a pre-amplifier, high-voltage supply unit, scaler and timer, the last two being provided with digital read-out for continuous elapsed time and accumulated count data.

The pre-amplifier, for use with Geiger or



Right: The Nuclear-Chicago Logic 801.

scintillation counters will accept either positive or negative pulses, and provides gains of 1 or 20. The high-voltage unit gives continuous variation from 350 to 5 000 V positive, and 350 to 1 500 V negative, with an output regulation of 20 ppm/mA at 1 000 V. The scaler will accept either positive or negative pulses, and has a resolution of 1 μsec. The timer will indicate in increments of 1 msec with a maximum drift of 1 ppm/week, and is capable of feeding a digital recorder. Decimal point setting is provided by the range switch.

(Nuclear-Chicago Corporation, 359 East Howard Avenue, Des Plaines, Illinois. Continental Distributors, Ltd., 121 Earls Court Road, London, S.W.5.)

1140

Hygrometer Alarm

Continuous indication of moisture and alarm or control contacts acting at any desired point on the scale are features of a new ultra-sensitive hygrometer

Shaw ultra-sensitive hygrometer.

developed by Shaw Moisture Meters, from an indicating instrument that has been widely in use for over two years in situations where water vapour, in microscopically small amounts, could cause trouble. The instrument will operate successfully when immersed in liquids, and is said to be capable of giving full-scale deflection when the sensing head is immersed in kerosene containing 50 ppm of water.

The instrument is scaled in dew-point temperature (−120°C to −20°C) or in % R.H. as desired and can be located several hundred yards away from the sensing head, to which it is connected by a coaxial cable. Several sensing elements can be operated from one instrument if desired. Maximum temperature rating is 180°C.

The sensing head is protected by a sintered bronze sheath, giving a construction sufficiently robust to be suitable for industrial operation.

(Shaw Moisture Meters, Rawson Road, Westgate, Bradford.)

1141

Pressure Spray Equipment

The convenience of pressurized aerosol containment and the capability of compressor operated spray equipment are combined in the new

Solus-Schall "Flexi-Spray."

Flexi-Spray unit for penetrants, magnetic inks, and similar materials. While the standard type of aerosol pack is very convenient for many purposes, it is at a disadvantage with liquids of high viscosity or with those carrying solids in suspension, which tend to clog the valve.

In the Flexi-Spray, the propellent gas and the spray liquid are contained in separate units, both of which fix into a nozzle unit of moulded plastics, which operates in the same manner as a normal low-pressure gun, drawing the liquid from the container by a vacuum created over the top of the feed pipe. The propellent unit weighs 14 oz and is capable of spraying about 20 fl oz of the kerosene used as a vehicle for magnetic materials.

(Solus-Schall, Ltd., County Building, Honeypot Lane, Stanmore, Middlesex.)

1142

Automatic Welding

Although a non-nuclear application, the accompanying illustration of automatic welding on large tanks at the Purfleet site of Shell-Mex and B.P. Ltd., shows the capability of the "Arcos" Circomatic unit for circumferential seam welding. Running in monorail fashion on the upper edge of the course, this 30-cwt machine automatically welds from both sides, plate up to ¾ in thick being welded in a single encirclement of the tank.

Setting-up and welding times are said to be only a fraction of those for manual welding, and it is stated by Whessoe, the contractors for this site, that a welding speed of 24 in/min is now regarded as normal for ¼ in plate. Gas-shielded arcs and "Comet" flux-cored wires are used.

(Rockwell, Ltd., Commerce Way, Croydon, Surrey.)

1143

Rockweld Circomatic unit working on circumferential seams of a large tank.

Laboratory Irradiation Units

Laboratory units for cobalt-60 irradiation, manufactured by Atomic Energy of Canada, Ltd., were described in *Nuclear Engineering* for January, 1960, p. 43. Two new units are now announced, the Gammacells 100 and 200. As in the previous model, there is a fixed ring of encapsulated rods, into which the specimen is lowered. The cell for the 100 model is 1½ in diameter x 4½ in high; for the 200, the corresponding dimensions are 3½ in diameter x 5½ in high. The sample

BRIEFLY . . .

The Power Flexible Tubing Co., Ltd., of Derby Works, Vale Road, London, N.4, are marketing "Jetflex" corrugated flexible tubing in titanium-stabilized stainless steel, with an external braiding of stainless steel wire to eliminate elongation under pressure, and suitable for welding to stainless steel end fittings. Development has entailed an extensive test programme, including fatigue tests under cyclic bending and pulsating pressures; design pressures up to 2 000 psi are possible. **B293**

A new ultra-high vacuum pump is announced by **Mullard, Ltd.,** Mullard House, Torrington Place, London, W.C.1. Known as the VPP-5, it operates on the principle of the Penning gauge, using a large number of electrode units in parallel. Pumping begins at 5×10^{-3} mmHg and will continue at 5 litres/sec (air) to a pressure of 10^{-11} mmHg. The maximum operating temperature is 250°C; bake-out temperatures of twice this value may be applied. **B294**

Rocol, Ltd., Rocol House, Swillington, near Leeds, have introduced Molydite Plus, a thermo-setting varnish for production of a resin-bonded lubricating film of molybdenum disulphide. This gives better results (i.e., a more durable film) than their previous product, Molytox Plus, but requires more care in application and heat treatment. Molytox Plus, which was used on the control mechanisms at Calder, is being retained for general-purpose use. **B295**

"Clamps," or soft jaws for holding work in a vice without marking or damage, are no novelty, but a new twist is given by **Speed Tools, Ltd.,** Vereker House, Gresse Street, London, W.1, whose clamps are made of tough plastics material, with inset permanent magnets for instant positioning in the vice. The jaws being thermo-plastic, a heated component will make an impression that can act as a permanent jig, if required. **B296**

R. H. Lighting, Ltd., Anyards Road, Cobham, Surrey, are distributing Metalphoto nameplates, a recent U.S. development. Photosensitized aluminium plates may be printed from normal negatives or tracings (either line or half-tone) and developed in the same way as a bromide print. The resultant image, after 30 min in boiling water, is covered by a layer of oxide 0.0005 in thick which, although perfectly transparent, is impervious to scratching or peeling, salt spray, organic solvents, or temperatures up to 1 000°F. A rapid processing service is available, if required. **B297**

A new range of sectional prefabricated buildings, the SB-2, is announced by **Middlesex Prefabricated Buildings, Ltd.,** Hounslow, Middlesex, with an outer cladding of resin-bonded plywood, in place of weatherboard, thus eliminating leakage and draughts. **B298**

The U.S. associates of **Honeywell Controls, Ltd.,** Greenford, Middlesex, have developed a high-density magnesium oxide (within 2% of theoretical density) which is translucent and capable of taking a high polish. Its compressive strength is in the region of 45 000 psi as against 24 000 psi for ordinary MgO, and it has a melting point of 2 800°C. **B299**

General Precision Inc., 63 Bedford Road, Pleasantville, New York, have developed a combination of microfilm read-out and closed circuit television. A microfilm card placed in a transmitter at, say, a central library, may be examined at a remote monitor where full facilities are available for examining any portion of the original and enlarging detail up to 1 000 times at will by a simple "joystick" control switch. **B300**

Automatic voltage stabilizers in large ratings (up to 120 kVA eventually) are being developed by **Claude Lyons, Ltd.,** Valley Works, Ware Road, Hoddesdon, Herts. The first of the range to be placed on the market, the TS-1, has a rating of 1 kVA with 25% variation of the input voltage, or 2 kVA if the input does not vary by more than 12½%. A buck-and-boost transformer in series with the mains is energized from a motor-driven Variac positioned by a transistorized sensor unit, or servo-amplifier, which monitors a rectified sample of the output voltage against a zener diode. The accuracy is said to be better than 0.5% from zero to full load, and the response speed 40 V/sec. **B301**

Interlas, Ltd., Ampthill, Bedford, announce a new range of a.c. welders known as the Unitor, in sizes of 200, 300, 400, 500 and 600 amp, with infinitely variable output. Three-operator sets can also be supplied. All can be provided with wheels and pulling handles for easy mobility if desired, the three larger sizes can be wall-mounted, and provided with the patented Unitor current control whereby any operator may vary his current from the welding position without the use of auxiliary cables. **B302**

The Gammacell 200 laboratory irradiation unit.

chamber is lowered into the active area by electric motor, under the control of a preset timer. Additional experiments can be accommodated in a helical tube surrounding the source.

Irradiation levels in the centre of the chamber may be as high as 3×10^6 r/h; the shielding is such that the external doserate is less than 5 mr/h at 12 in from the surface, using the maximum source—no additional shielding or special enclosure is required.

(Watson and Sons (Electro-Medical), Ltd., East Lane, North Wembley, Middlesex.)

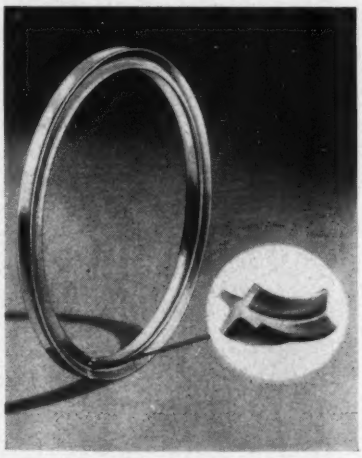
1144

X-ring Seals

The Wiggins BAR-X metal sealing ring has been developed for application to a wide range of static and dynamic sealing problems and is particularly successful with many hot gases and exotic fluids, in fact the claim is made that "if metal will hold it, BAR-X will seal it." The general service limitations are given as −455°F to 1 200°F with pressures up to 6 000 psi. The standard range extends from ¼ in i.d. to 3 in i.d. for static seals, or 2 in for dynamic sealing, although any size required can be supplied. The seal is stated to be geometrically stable, being unaffected by direction of motion or pressure. It can be supplied in a variety of hard-surface materials for wear resistance.

(Aero Controls, Ltd., Industrial Estate, Weedon Road, Northampton.)

1145



The BAR-X metal sealing ring.

In our description of the PNI 1091 Ionization Chamber (*Nuclear Engineering*, August, 1961, p. 354) we failed to make it quite clear that the chamber itself was developed by Plessey Nucleonics Ltd., and the filling gas by AERE, Harwell.

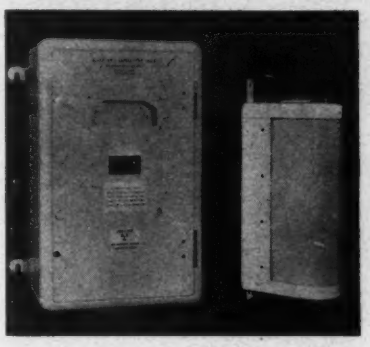
Gamma Radiation Monitor

The type 1641 C radiation monitor is now being manufactured under AEA licence and is commercially available. Intended for laboratory areas and the like, this instrument consists of a detector and control unit giving continuous indication of radiation levels, with a 3-colour (green, amber, red) display unit and a warning bell which rings continuously or intermittently according to the level.

The detector and control unit houses a 3·5-litre aluminium ion chamber with a low-energy priming source, and the necessary electrometer and control circuits. The electrometer circuit feeds a moving-coil instrument calibrated from 0-300 mr/h. A blade on the pointer intercepts the light to either of two phototransistors to operate alarm relays, the normal settings being 30 mr/h and 285 mr/h; these settings are readily adjustable.

The detector unit contains a calibrated strontium-90 source which can be rotated into position, against a spring, in front of an aperture in the ion chamber, for test purposes.

The unit operates from a.c. mains. All indicating lamps are duplicated and under-run for reliability; the design of the lamp unit gives good visibility through 180° in daylight. Provision is made for operation of additional remote lamp and bell units.



Redcliffe 1641 C gamma radiation monitor.

(Redcliffe Radio and Engineering Co., Ltd., Emery Road, Brislington, Bristol, 4.)

1146

CATALOGUES

Nos. 20 and 21 of the *SS News* published by **Solus-Schall, Ltd.,** County Building, Honeypot Lane, Stanmore, Middlesex, contain an article on non-destructive techniques for assessing radiation damage in pressure vessel structural materials, by Dr. Friedrich Forster.

"Tollcross Steel Foundry" is the title of a brochure published by **Stewarts and Lloyds, Ltd.,** Brook House, Upper Brook Street, London, W.1. This gives a brief review of the processes involved in the manufacture and testing of steel castings, a glance at some typical products, and lists of standard- and special-purpose irons and steels, current publications, and addresses.

The Marconi Type TF 1299 Radiation Monitor is described in Vol. 8 No. 2 of *Marconi Instrumentation*, published by **Marconi Instruments, Ltd.,** St. Albans, Herts.

The International Rectifier Co. (Great Britain), Ltd., Oxted, Surrey, have issued a new short form catalogue of products.

Tyer and Co., Ltd., Perram Works, Merrow Siding, Guildford, Surrey, have issued a leaflet on the standard range of Perram transistorized photo-electric counters.

Kennametal Inc., Latrobe, Pennsylvania, U.S.A., have issued a booklet on Kennertium W-10 heavy tungsten alloy.

Johnson, Matthey and Co., Ltd., 73-83 Hatton Garden, London, E.C.1, have issued a new series of data sheets on their range of small-bore tubes and tube products in non-ferrous metals, nickel-iron alloy, and precious metals. Tubes are listed in terms of end uses, e.g. Bourdon, capillary, restrictor, instrument pointer, etc.

A new list from **Cambridge Instrument Co., Ltd.,** 13 Grosvenor Place, London S.W.1, describes their indicating pneumatic temperature controller for operation between −20°C (10°F) and 340°C (650°F).

The twelfth edition of "Electrostatic Precipitation" has been received from the **Sturtevant Engineering Co., Ltd.,** Southern House, Cannon Street, London, E.C.4. Other Sturtevant lists received cover paddle-bladed D fans and "Stu-te-pak" anodes for cathodic protection of pipes and structures.

Leaflet SL.I covering British Standard totally enclosed fan-cooled squirrel-cage motors has been received from **Lancashire Dynamo and Crypto, Ltd.,** Trafford Park, Manchester 17.

The Federation of British Industries, 21 Tothill Street, London, S.W.1, have issued a handbook "Height and Design of New Chimneys."

The complete range of nuclear instrumentation services (design, development, manufacture, installation, maintenance and testing) is described in bulletin GEA-7259 received from the **General Electric Company,** Schenectady 5, New York.

"Moment of Decision," a book received from **Hayward Tyler and Co., Ltd.,** P.O. Box 2, Luton, Beds, describes the Hayward Tyler apprentice training scheme.

"Gamma Irradiation in Canada" is the title of a publication issued by **Atomic Energy of Canada, Ltd.** (Commercial Products Division). In addition to information on dosimetry and radiation facilities, there is a great deal of information on the preservation of foods, in particular, irradiation of potatoes. A limited number of copies are available from **Watson and Sons (Electro-Medical), Ltd.,** Industrial Division, East Lane, North Wembley, Middx.

The Heavy Organic Chemicals Division of **Imperial Chemical Industries, Ltd.,** Millbank, London, S.W.1, have issued a brochure on "Thermex" heat transfer medium, a diphenyl oxide-diphenyl eutectic with an atmospheric boiling point of 495°F (257°C).

Oxygen analysis equipment is described in a new specification sheet received from **George Kent, Ltd.,** Luton, Beds.

The Admel precision drafting machine is described in a brochure received from **Lawes Rabjohns, Ltd.,** Abbey House, Victoria Street, London, S.W.1.

"The Allen West Story" commemorating 50 years of electric control gear equipment, has been received from **Allen West and Co., Ltd.,** Brighton, 7.

A new machinery catalogue has been issued by **Thos. W. Ward, Ltd.,** Albion Works, Sheffield. It is divided into seven sections: machine tools; sheet metal working machinery; woodworking machinery; small tools; power plant; foundry and forge plant; electrical plant. The first issue of a 16-page stock list, to be produced monthly, has also appeared.

Alma Components, Ltd., 551 Holloway Road, London, N.19, have issued a quick reference wall-chart of precision wirewound resistors.

The Computer Department of **Ferranti, Ltd.,** 68 Newman Street, London, W.1, have issued a new classified index of Ferranti computer literature.

Orbits . . .

REGULAR READERS—both of them—are invited, nay, requested to consult their bound volumes (or the bottom of that tattered heap in the corner) and read these notes for February, 1957. For those tidy souls who do not keep back numbers—and wouldn't bother to look even if they had—we extract the relevant paragraph, referring to the sod-cutting ceremony at Bradwell:

> It was a bold challenge that Sir Claude Gibb flung to his competitors, incidentally, in prophesying that Bradwell would be the first truly economical station in the world to go into regular service. This is no sprint, where the result is decided by the competitors' first leap as the pistol bangs; it is more of a five-miler, where almost anything can happen.
>
> If there really is a race, it should provide all the makings of a really good film plot. We should like to see the way Hollywood might handle the scenario. One can picture the goings-on between rival organizations; the masked figures creeping on to the sites at dead of night to cut crane cables; the fake fuel elements full of dynamite; the final scene where the chairmen of the respective companies fight it out, man to man, with rolled umbrellas on the jib of a crane high above the site . . . but real life is much less colourful, unfortunately.

To our lasting regret, Sir Claude is no longer with us—and the engineering world is the poorer for his passing. Speculation regarding the course of events had he lived is fruitless, and the so-called race has almost evaporated. Hunterston, without the ill-luck that dogged the GEC, was delayed, for nearly a year while committees wrangled over the choice of site; Berkeley and Bradwell are now under the same ownership; and even Hollywood would be hard put to concoct a plot for sabotaging one's own property. Besides, there was no real need for sabotage so long as our old friend the CID (Change in Design) was busy; graphite, not dynamite, did the trick.

Still Hope

Both stations, we hope can now be considered on the last lap. Round about the end of July, it appeared that Berkeley had a slight lead, as it was reported ready for loading earlier. Nevertheless, loading started within a couple of days of each other—Berkeley on August 12 and Bradwell on August 14 and, being a faster loader, Bradwell went critical first. So, unless the owners exercise the prerogative of racehorse owners, and hold one back having declared to win with the other, it seems to be still anybody's race. The original prophecy might still come true.

Not Always to the Swift . . .

Ectuallay, as they say in the Navy, both the CEGB and the Min. of P. greet the idea of a "race" with the deepest antipathy, because it tends to create in the public mind (for want of a better term) an impression that both sides are pressing on regardless of anything else but speed. So perhaps we'd better lay off all mention of winner or loser, and regard the matter in the light of an exhibition spar.

Crossing Coincidence

It was rather a coincidence that Sir Christopher Hinton, lured on to the platform for a B.B.C. "Listeners Answer Back" programme, should have referred to the 400 kV Thames crossing at West Thurrock within a couple of days of the Press visit organized by B.I.-Callender's to the Swanscombe side of the river. Sir Christopher was answering questions on the power failure in the south-east area earlier this summer and, although he made only the briefest reference to the crossing, it was sufficient to show its importance to the maintenance of electricity supplies in this country.

Though not, as Gracie Fields used to sing, the "Biggest Aspidistra in the World," the West Thurrock crossing is a remarkable piece of engineering, and must have provided a good many headaches for its designers—not least of which were the twin requirements that the conductors must not at any time touch either the water or the ground, neither was any clamping of the conductors permitted, during erection. When one is dealing with 400 kV, a smooth surface is essential, unless one wants a corona glow that would put Cerenkov radiation in the shade.

Added to these were a few other details which didn't help very much. The ground on the banks of the Thames is hardly ideal for foundations (each main tower needed 30 piles to each leg); the PLA required a minimum of 250 ft clearance to the water at maximum sag; the total river closure permitted for stringing the haulage rope was four hours; the weight of each of the six 7 754-ft conductors was 21¼ tons—it doesn't get any easier, does it? We always thought that the 1932 crossing at Dagenham (also, incidentally, a Callender's job) was pretty impressive; the West Thurrock crossing makes it look like a radio amateur's transmitting aerial, by comparison.

Nuclear Red Herrings

It is some time, incidentally since we heard Sir Christopher speak, but he does not seem to have lost his touch with an audience. After once making them laugh over the precise definition of "gross negligence" ("gross negligence is something that we are never guilty of") it was a simple matter for the Hinton-Elliott team to kick around, and, finally, lose a snorter like "How long will it be before sufficient atomic power is being generated to make off-peak tariffs cheaper?" In very little time we were off, via underfloor heating and storage hot water, to a much safer question, on how the CEGB is handicapped by the wayleaves situation, and how a recalcitrant landowner can cost them thousands of pounds. That dirty word "coal" was hardly mentioned.

Brief Encounter?

According to a recent BBC news item, the first job for the new Atlas computer for NIRNS is ". . . to investigate the complex behaviour of nuclear reactors."

After it has flicked aside that small chore, it will, presumably, be available for something really difficult.

Affluence in Effluence?

The leak in the Harwell effluent line at Sutton Courtenay must have been a sad disappointment to many popular newspapers, who haven't had a good Atom Disaster story for some time (although, as we write, we note that the *Daily Skreetch* has invented a Peril, which is the next best thing). The Sutton Courtenay incident was so trivial that the only memorable comment came from a "spokesman" (presumably an itinerant bicycle repairer) who stated that it would have been safe to drink a gallon of the sludge. We have waited patiently since early August for further developments, without results.

This negative attitude in public relations is to be deplored. It is not sufficient to show that it was harmless, it should have been proved beyond all doubt that it was beneficial. There should be front-page pictures of Cabinet Ministers (or, better still, pop singers) actually drinking the stuff; of glamorous (?) models, with hair like Guardsmen's bearskins and eyes like post-holes in the snow, taking mud baths. Properly handled, Sutton Courtenay could receive such a build-up that it would eclipse Buxton or Bath; there would have to be a permanent hole in the pipe to supply the demand; the AEA would find that mud, not isotopes, was the real money-spinner. It's too easy . . .

Come to Sutton Courtenay
Where the healing waters flood.
With the help of AEA
Bathe in radioactive mud;
Come to Sutton Courtenay
Britain's new health-giving spa,
Where the magic Glamour Ray
Cures coryza or catarrh;
For complexion like a peach
See how wonderful it is.
Let your epidermis bleach
With a mud-pack on your phiz;
Take the necessary steps
If your skin looks dull and dead.
Mud and milli-microreps
Soon will banish Washday Red . . .
Come to Sutton Courtenay
DRINKA . . .
PINTA . . .
SLUDGA . . .
DAY!

Target